NUCLEAR SAFETY

A REVIEW OF RECENT DEVELOPMENTS

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foreword

This quarterly journal is a Technical Progress Review prepared by the Oak Ridge National Laboratory at the request of the Office of Technical Information, U. S. Atomic Energy Commission. This Review is intended to assist those interested in keeping abreast of significant developments in the field of nuclear safety. *Nuclear Safety* is not a comprehensive abstract of all literature published in this field during a given quarter; rather it is a mechanism for presenting concise reviews of selected subjects as prevailing interest and available information warrant.

Coverage of the Review is limited to topics relevant to the analysis and control of hazards associated with nuclear reactors, operations involving fissionable materials, and the products of nuclear fission. Primary emphasis is on safety in reactor design, construction, and operation; however, safety considerations in reactor fuel fabrication, spent-fuel processing, nuclear waste disposal, and related operations are also treated. Safety in the use of radioisotopes in industry, medicine, and research is excluded, as are most topics considered to be the province of health physics. Even with these exclusions, nuclear safety overlaps such diverse fields as nuclear physics, solid-state physics, mechanics, chemistry, meteorology, geology, seismology, metallurgy, law, and nearly all branches of engineering. The authors will therefore review material from these fields which, in their opinion, has a direct bearing on nuclear safety.

Two distinctly different types of articles are in this issue of *Nuclear Safety:* reviews of current literature and special review articles on specific topics. The editors feel that each of these articles makes a necessary and distinctive contribution to this journal. The special review articles permit discussion of pertinent subjects which cannot be adequately considered by reference to only the current literature. The current review articles, however, constitute the major portion of this issue. All incoming literature (including reports, books, American and foreign technical journals, and transactions) is examined for subjects within our area of interest. This material is collected, grouped, and reviewed by experts. Interpretations, in any article, represent the opinions of the editors, who are employees of the Oak Ridge National Laboratory. Readers are urged to consult the references to original work for more complete information.

It is recognized that the critical evaluation of subject areas leading to the determination of criteria cannot fail to stimulate contrary opinions. This is expected to be particularly true in the area of nuclear safety since, in many instances, only preliminary information is available, the ramifications are many and varied, and opinion and judgment must be relied upon so heavily. Although the editors do not propose that *Nuclear Safety* act as a clearing house for safety correspondence because of the above facts, we have had for some time a policy that would permit the publication of statements of position at variance with those expressed by the editors. Such statements will be published after the editors have ascertained that a real difference exists and that the position is reasonable.

In addition to the invited contributors, many members of the Oak Ridge National Laboratory staff wrote review material, reviewed manuscripts, or otherwise contributed to this publication. Their contributions are gratefully acknowledged.

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Section 1

GENERAL SAFETY CRITERIA

Reactor Siting Trends and Developments

Questions of reactor hazards and the adequacy of their evaluation are increasingly important in the atomic energy program. They affect its costs, its progress, and its acceptance by the public. The viewpoint of the citizenry continues to be strongly affected by the public press, whose writers are well aware of the news value of each and every "atomic incident." Hence the deaths of three soldiers in an automobile crash receives scant and transient notice compared with the deaths of three soldiers in a reactor accident.

It is not surprising that the increase in the number of operating reactors and the increase of the exposure hours as the years pass are combining to yield statistics that eventually will be the basis for better estimates of risk. The public interest in reactor safety is wholesome and proper, particularly on the part of the persons in the environs of a proposed reactor site. It is likewise proper for the reactor operator to undertake the public-relations task of generating a more informed public opinion so that those concerned can better appreciate the problem.

As mentioned in an earlier issue of *Nuclear Safety*,[1] the proposition of establishing some criteria for the siting of reactors is of concern to many groups, including the American nuclear industry, health authorities, and the general public as well as the AEC. The AEC, having the statutory responsibility for licensing the operation of reactors, "requires reasonable assurance that the health and safety of the public will not be unduly endangered. [2] The safety of specific reactors proposed for location at particular sites has been, from the outset, the subject of careful scrutiny, often by the ACRS and in every case by the AEC's Division of Licensing and Regulation.

There appears to have been a very definite increase in emphasis on the part of the licensing authorities in the past few years to make more definitive the basis on which reactor sites are evaluated and to minimize dependence on subjective judgment. This attempt to simplify the complex problem of reactor hazards and site evaluation at least to the extent of formulating guides for the selection of sites is for the purpose of giving reactor planners as definite information as possible to use in estimating costs. Although "subjective judgment" in hazard evaluation cannot be avoided, the trend toward the resultant more cut-and-dried viewpoint is readily seen.

In a speech by Beck[3] at Rome, June 1958, the point was well made that:

> Basically, the biggest variable in the choice of sites for reactors is the reactor itself. A given location might be quite acceptable for one reactor but not at all acceptable for another reactor. Therefore evaluation of sites for reactors must begin with evaluation of reactors for which sites are sought.

Beck also clearly stated that, if the potential exposure doses were found to be above those which are acceptable, the reactor facility might have to be modified with respect to shielding or containment. As a tentative value for calculated maximum emergency doses, 25 rem to the whole body was suggested. Here, recognition is given to the fact that the potential hazard depends on both the quantity and identity of the released fission products.

Somewhat later, at Geneva, in September 1958, essentially the same point was made,[4] perhaps even stronger:

> The important features of the proposed site must be considered in applying these criteria, but it is

> equally important that the features of the reactor and of its associated equipment be considered. Thus, the reactor and the site must be evaluated together, and it follows that certain safety deficiencies or inadequacies in either the reactor or the site can be offset in part by compensating characteristics in the other.

This Geneva paper also contained several excellent statements with regard to inspection, among these being that over-all safety of reactor operation requires periodic observations and review during normal operation.

In May 1959, the AEC published for public comment proposed regulations on site selection, as discussed in an earlier issue of this Review.[1] These proposed regulations were broad and quite general in nature. Variables such as the inherent safety of various types of reactors and the character of the released fission products or of emergency exposure levels were not dealt with in any definitive manner; rather criteria were expressed in such general statements as, "It is usually desirable that the reactor should be several miles distant from the nearest town or city and, for large reactors, a distance of 10 to 20 miles from large cities." Upon publication of the proposed regulation, there was widespread interest, some favorable comment, and a number of protests. The development of thoughtful study of the problems has been one good result. Some of the comments have been discussed heretofore,[1] and many others were brought out at the Technical Appraisal Task Force meetings convened by the Atomic Industrial Forum[5] on June 1 and June 30, 1959. The controversy served to focus attention on the complexities of the problem and further the interest of such groups as the ASA and the American Nuclear Society (ANS), both of which have active committees to study site problems.[6–8] Unfortunately, there were some who saw in the proposed regulations further support for their misgivings and alarm about any and all reactors. Furthermore, a strict interpretation of the regulation could have worked an injustice on the reactor designer and manufacturer (inasmuch as the proposed regulations did not make allowances for efforts toward increased design safety) and on the energy user (who could have been required to accept, in some cases, unnecessarily expensive transmission costs).

Also, at Rome, in June 1959, the need for analysis of an entire reactor proposal before judging its acceptability was presented:[9]

> It is immediately obvious that this [nuclear reactor accident] hazard can be avoided only by evaluation and definition of suitable specifications for both the reactor under consideration and the site proposed for its location. Neither can be properly evaluated without the other.

At this time the credibility of accidents was discussed, and it was concluded that safeguards can be provided against any credible accident. The worst conceivable accidents, requiring the failure of all these safeguards, were recognized to be unrealistic. It was pointed out that the systematic search for credible accidents often contributes substantially to the safety of a facility. In order to illustrate the safeguards provided, containment was discussed as one of the "factors of crucial importance" in the selection of a site for a reactor. It was reasserted that, before approval, each project must be evaluated separately and a decision reached that the combination of the reactor facility and the site will not offer an undue hazard to people in surrounding areas. It was concluded that it was

> ... not possible to define such [reactor site selection] criteria without utilizing arbitrary and controversial rules.

The paper also described

> ... generally the choices which have been made in the past, though the pattern has not exhibited a great deal of consistency.... These statements can be employed, however, only as general guides from past practice, and each new project must be comprehensively examined by practices which remain at present largely subjective.

It must be appreciated that such summaries, based on past practice, reflect, at least in part, the conservatism necessary when reactors were wholly new and untried.

Although repeatedly pointed out as a problem that must be considered as a unified whole in order to be considered adequately at all, efforts to deal exhaustively with separate aspects of site selection continue. For example, ASA has a variety of separate committees and subcommittees concerned with site selection, containment, credible accidents, etc.,[7] which are related but which are considered by the committees as totally separated topics. It has been particularly well pointed out by Binford in a recent review in this Review[10] that the evaluation of a hazard involves not only the detailed

consideration of the consequences of a nuclear accident, both personnelwise and costwise, but also the assessment of the risk of the occurrence and the means available to reduce the risk or lower the probability of the accident.

The transactions of the Winter Meeting of the ANS in December 1960 contain a brief paper[11] by C. K. Beck, Assistant Director of the AEC Division of Licensing and Regulation, on site criteria for power reactors which describes a procedure that will give "acceptable" values for three particular distance characteristics required of sites for power reactors. These three distance characteristics are (1) an exclusive area immediately around the reactor under full control of the reactor operator, (2) a secondary buffer or evacuation zone in which the population density is relatively low, and (3) substantial isolation from a major city or population center. Unfortunately, these distance characteristics are given with no distinction as regards other equally important factors, such as reactor type and safety features.

Beck's paper was derived from the results of a study by the Division of Licensing and Regulation. The results of this study as subsequently issued as a proposed regulation[12] elaborated further on the applicability of the criteria as follows: "it is not possible to define such criteria with sufficient definiteness to eliminate the exercise of judgment in the evaluation of these sites" and that it "is intended primarily to identify a number of factors considered by the Commission." Calculations in an appendix to the proposed guide are included as:

> ... a means of obtaining preliminary guidance. They may be used as a point of departure for consideration of particular site requirements which may result from evaluations of the particular characteristics of the reactor, its purpose, method of operation, and site involved.

Listed in this proposed guide are the following factors that are to be considered in determining the acceptability of a site for a power or testing reactor:

> (a) Population density and use characteristics of the site environs...
>
> (b) Physical characteristics of the site, including, among other things, seismology, meteorology, geology, and hydrology...
>
> (c) Characteristics of the proposed reactor, including proposed maximum power level, use of the facility, the extent to which the design of the facility incorporates well proven engineering standards, and the extent to which the reactor incorporates unique or unusual features have a significant bearing on the probability or consequences of accidental releases of radioactive material.

The proposed exclusion area (that under full control of the reactor licensee), low population zone (area immediately surrounding the exclusion area whose residents may be subjected to appropriate protective measures), and the population center distance (distance to nearest boundary of a densely populated center of more than 25,000 residents) were to be determined as follows:

> (1) An exclusion area of such size that an individual located at any point on its boundary for two hours immediately following onset of the postulated fission product release would not receive a total radiation dose to the whole body in excess of 25 rem or a total radiation dose in excess of 300 rem to the thyroid from iodine exposure.
>
> (2) A low population zone of such size that an individual located at any point on its outer boundary who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage) would not receive a total radiation dose to the whole body in excess of 25 rem or a total radiation dose in excess of 300 rem to the thyroid from iodine exposure.
>
> (3) A population center distance of at least $1\frac{1}{3}$ times the distance from the reactor to the outer boundary of the low population zone. In applying this guide due consideration should be given to the population distribution within the population center. Where very large cities are involved, a greater distance may be necessary because of total integrated population dose considerations.

The proposed site criteria guide was published in the *Federal Register* on Feb. 11, 1961, and interested persons were invited to submit written comments or suggestions to the USAEC within 120 days after publication. The comments subsequently received will be reviewed in the next issue of *Nuclear Safety*.

(T. J. Burnett)

Safety in Nuclear Ship Propulsion

The proposition of nuclear ship propulsion safety has received considerable attention at both national and international levels during the past year. The feats of the U. S. nuclear submarines and the launching of the U.S.S.R. Ice-

breaker *Lenin* ushered in the age of nuclear vessels and directed attention to the capabilities of these vessels. It is, however, the advent of the N.S. *Savannah*, which is expected to ply the trade routes of the world and to dock at major port facilities, that has brought the proposition of nuclear ship safety to the international conference table. This past year has seen both the International Conference on the Safety of Life at Sea (SOLAS) in London and an IAEA symposium on nuclear ship safety in Sicily, as well as numerous articles on the subject. A previous article[13] in *Nuclear Safety* discussed the safety features of the N.S. *Savannah* and some related work in this country; this article is addressed to the broader implications of nuclear ship propulsion safety as it evolves at the international level.

In reality the additional considerations introduced into the problem of nuclear safety by a maritime environment extend and modify the problem, but they do not basically change it. Thus, if there were detailed understanding and agreement on the safety problems related to stationary power reactors, it could be expected that the additional problem of nuclear ship safety could be readily resolved. Such is not yet the case, and the additional considerations that occur in connection with nuclear ships compound the problems that are still to be resolved for their stationary predecessors. Although this point needs no elaboration, it is most appropriately illustrated in this country by the difficulty in establishing reactor site criteria (see the preceding article in this issue) and the fact that the licensing procedure has been challenged.[14]

In May and June 1960, an international conference was held in London under the sponsorship of the Intergovernmental Maritime Consultative Organization (IMCO), a United Nations agency, to consider revision of the 1948 International Convention for the Safety of Life at Sea. Inasmuch as the previous convention made no mention of nuclear vessels, one of the principal tasks of the 1960 convention was to determine requirements, in so far as possible, for nuclear ships. Delegates of more than 60 nations attended the convention. The positions of the British[15] and American[16] delegates were documented in reports based on studies undertaken in preparation for that meeting. Both statements were deliberately nonspecific, since it was generally believed that it was too early in the development of nuclear ships to include any detailed safety requirements in an international convention.[17,18] The British recommendations have been separately discussed in an article by Neuman.[19]

It is not surprising that some of the recommendations[20] of the SOLAS conference not only bear a marked similarity to the prior studies but also are neither as specific nor as restrictive as either the U. S. or British recommendations. Of singular importance is the omission from all the recommendations of a specific definition of the extent to which any radiation exposure, either accidental or routine, must be controlled. Thus, for example, although all recommendations are quite concerned with the various features of containment vessels, there is no quantitative definition of the function of this vessel. Until the extent to which the radiation dosages resulting from an accident must be controlled is established, it will not be possible to evaluate the adequacy of containment vessels or other containment schemes. The recommendations do, however, identify important areas of concern regarding nuclear safety, and these will be further elaborated upon at future meetings. It should hardly be expected that an international convention could resolve questions that have not yet been resolved on national levels.

In November 1960, the IAEA (in cooperation with IMCO) sponsored the symposium on Nuclear Ship Propulsion with Special Reference to Nuclear Safety at Taormina, Sicily.[21] This meeting was attended by approximately 175 delegates representing 19 countries and six international agencies. The papers presented at the meeting included many on specific ship reactor designs and safety problems. In general, these were presented in the light of the safety concepts developed at the SOLAS conference and did not attempt to go any further in this area. The following specific conclusions derived from the Sicily meeting were summarized by H. Whiteside of IMCO, who presided over the closing session:

1. Nuclear ships may be built and operated to acceptable standards of safety.

2. The meeting brought to light many problems and openly discussed them.

3. Any nuclear merchant ship built in the next few years will be uneconomical.

4. Experience on actual nuclear ships is needed.

5. Detailed rules and regulations that might restrict new designs and developments are to be avoided.

Although the papers presented at this conference covered all aspects of nuclear ship propulsion, including economics, international problems, various ship designs, shielding, maintenance, and refueling, as well as safety, two papers were of particular interest because of their attempt to define in quantitative terms the safety considerations involved in the operation of nuclear ships. In both these papers the authors attempted to define the extent to which the consequences of an accidental release must be controlled. Bell[22] suggested a limiting individual exposure value of 25 rem to the thyroid, implying that this could result from a release of the order of tens of curies under some circumstances. Cottrell[23] suggested a value of 25 rem to the whole body or its iodine equivalent (given as 67 rem to the thyroid), in addition to a restriction on the total population exposure, which was placed at 2×10^6 man-rem to the whole body. [See *Nuclear Safety*, 1(2): 2–5 (December 1959) for additional discussion on this point.]

It was announced by H. de Laboulaye of IAEA, who was Acting Director General for the symposium, that IAEA was considering the establishment of a nuclear ship safety group to assist member states in the evaluation of safety problems attendant to nuclear ship propulsion. This would be the counterpart of a service recently inaugurated by IAEA to assist member states in the evaluation of stationary power reactors.[24]

Although the use of international agencies for the evaluation of reactor hazards is a recent innovation, its virtue is not confined to the fact that smaller nations with limited technical resources may benefit from such competence. Even with stationary power reactors, the consequences of an accident or activity release may not be restricted to the boundaries of the country in which the facility is located, and the establishment of an international review agency is a step toward international control for the mutual protection of all concerned. Furthermore, with a mobile reactor designed to travel the international trade routes and to visit foreign ports, the ramifications of the safety requirements for that ship design unquestionably extend beyond the areas of national cognizance. At the present time, with the limited number of nuclear vessels and inadequately developed safety standards, the question of ship safety controls may be handled by bilateral agreements; however, the ultimate objective, as advocated in a paper by Hydeman and Berman,[25] should be a system of international control. According to these authors the advantages of such a system, aside from the reasonable freedom of operation it would afford nuclear shippers, would be that it would give assurance to maritime nations that nuclear vessels visiting their ports were not a hazard, it would preclude international disputes by the clarification of national rights, it would give protection against inadequate controls imposed by the present practice of sailing ships under "flags of convenience," and it would provide a basis for an adequate solution to the liability problems involved.

If, as may be inferred, it is intended to consider the safety aspects of nuclear ships as separate and distinct from those associated with stationary reactors, it would be of interest to examine the differences between the two. These were tabulated by Cottrell,[23] as follows, and are discussed below:

a) Proximity of nuclear installations to passengers as well as crew;

b) Varying size exclusion areas and surrounding population densities from one port to another;

c) Possibility of release to the hydrosphere;

d) Probability of accidents as a consequence of the mobility of the plant;

e) Possibility of failures as a consequence of the motion of the ship;

f) Emergency power must be provided by self-contained power sources because there are no external power sources;

g) Evacuation of ship not always feasible or less hazardous than radiation exposure;

h) Continued operation of the reactor may be desired even at some risk to the occupants of the ship;

i) Discontinuities in nuclear plant operation, i.e., reactor shutdown or operating at low power in harbors, and necessity of rapidly varying power levels for maneuvering at sea;

j) Minimum public hazard potential external to ship when on high seas.

The design of a nuclear vessel and its hazards analysis should take into account each of these differences. The crew of a nuclear ship is the counterpart of the operating staff of a stationary power plant. Crew members concerned with reactor op-

eration and maintenance belong in the category of radiation workers and would be subject to occupational exposure limits as defined by ICRP.[26] Passengers on nuclear ships may be regarded, for the time being, in the same light as "members of the public living in the neighborhood of controlled areas."[27] If the exclusion area for a nuclear-powered ship is then reduced to the confines of the ship to which the passengers do not have access, the potential exclusion areas provided by dockside facilities will be more than adequate, although the exposure to the surrounding population, following an accident, must be taken into account. Release to the hydrosphere would be of particular concern if the ship were to sink in confined waters. The other differences are of varying degrees of importance; items d, e, and f place additional restrictions on the design of a nuclear ship installation, whereas items g, h, i, and j do not appear to have a significant effect, although i and j would tend toward moderating the hazard potential. There is, as yet, little experience in the adaptation of these differences to nuclear ships, and most of that is centered at the present time about the N.S. *Savannah*.

If, as has been concluded above, we are unable to determine the implication of these differences in a safety evaluation, then for all practical purposes the standards for the evaluation of nuclear ships should be the same as those for stationary nuclear power plants. In instances in this country, this would indeed appear to be the case; however, the recently proposed reactor site criteria (see the preceding article in this issue) could preclude the proximity of nuclear reactors to populated areas. If these criteria are applied to nuclear ships—and there are no obvious reasons why these nuclear installations should be exceptions to such considerations—the criteria must allow safety features of nuclear ships to be relied upon to restrict population exposures in the event of an accident rather than upon separation distances which are unattainable in most ports. Although the proposed criteria in their preliminary form may not preclude such a possibility, it is by no means apparent what safety features would be considered adequate to permit a nuclear ship to enter a populated harbor area.

In a somewhat more general paper on marine reactor safety, Maddocks,[27] in addition to discussing design considerations for shielding, containment, and fuel-element integrity, has considered some of the hazards peculiar to commercial ships and made several safety recommendations for nuclear ships. The hazards listed are as follows:

1. Forces due to ship's motion.
2. Shock effects due to mines, etc.
3. Consequences of fire to external reactor systems.
4. Stranding as it affects shutdown heat removal.
5. Foundering as it affects area contamination.
6. Entry of nuclear-powered vessel into ports.
7. Availability of propulsion power under all conditions.

Unfortunately, Maddocks did not address himself to the requirements for safety in ports, although the paper contains several appropriate suggestions, as well as an analysis of shipping casualties over the last decade. With the exception of damage that might result from acts of war, it should be mandatory that all nuclear ships be adequately designed with respect to each of the other considerations, even as was the case with the N.S. *Savannah*.[13] Furthermore, these safety items were recognized in the recommendations of the SOLAS conference, although, as previously noted, the required provisions were not specified. (W. B. Cottrell)

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Section II

ACCIDENT ANALYSIS

Graphite Oxidation

High-temperature reactors which are inert gas cooled and which have a graphite moderator and/or graphite fuel elements are subject to a graphite fire hazard if a coolant line ruptures and air is admitted to the core. In order to evaluate the potential danger in such a situation, it is necessary to solve the material- and energy-balance equations describing temperature and gas composition variations within the reactor core following a set of hypothetical accident conditions.[1–3] The results of such calculations are not precise, however, because of the many approximations inherent in their development and the lack of accurate values for certain parameters in the equations.

A serious source of uncertainty in these calculations is the applicability of the data on the rate of the graphite-oxygen reaction as a function of temperature and other experimental conditions. This reaction has been studied by a number of investigators, but, because of its complex nature, widely different results have been reported. Further, comparison of the results of different investigators is quite laborious since rates are expressed in a wide variety of units and on a number of different bases; that is, mass of oxidizing sample, geometric surface of sample, internal surface (as measured by gas adsorption), and others. The proper choice of an oxidation-rate basis is not obvious, as has been noted in recent reviews of graphite-oxidation kinetics by Binford,[4] Walker et al.,[5] and the reviewer.[6]

Attempts to describe the rate of the graphite-oxygen reaction quantitatively are complicated by a number of factors, of which the following are the most serious:

1. The dependence of the reaction rate on oxygen partial pressure (reaction order) is in some doubt.

2. The primary reaction product may be either CO, CO_2, or both. Both products enter into secondary reactions, and it is difficult to determine which, indeed, is the primary product under any given set of conditions.

3. Since graphite is porous, reaction may occur on internal pore surfaces, as well as at the external surface. Hence, diffusion of oxygen and product gases within the pore structure may play an important role in determining the over-all rate of reaction. At higher temperatures, gas-phase diffusion of reactants to the external surface may also be a rate-determining mechanism.

4. The graphite-oxygen reaction is strongly catalyzed by traces of inorganic impurities. The quantitative effect of these impurities is difficult to determine since impurity content may vary within a given sample of graphite. Variations in graphite crystal orientation may also produce variations in oxidation rates.

5. Oxidation rates may be markedly accelerated by neutron and gamma radiation.

6. Rates may be retarded by the presence of certain halogen-containing substances in the gas phase.

The chemical combination of carbon and oxygen atoms is generally thought to occur by the following steps:[5] (1) adsorption of an oxygen molecule at an active site on the carbon surface, (2) formation of a carbon-oxygen complex and rearrangement of the complex to a readily desorbable product, and (3) desorption of the product gas (CO or CO_2). Any or all of these steps may influence the rate of reaction. According to the classical Langmuir theory,[7] however, if step 1 (adsorption) is much slower than

steps 2 and 3, the reaction will be first order in oxygen partial pressure, and if either step 2 or 3 is much slower than step 1, the rate will tend to be independent of oxygen pressure (zero order). Intermediate orders would be exhibited if the rates of these steps were of comparable magnitude.

Although the kinetics of the reactions of carbon with CO_2 and steam can be explained by such a mechanism, the carbon-oxygen reaction shows an important anomaly with respect to dependence of reaction order on oxygen partial pressure. At low oxygen pressures, where the adsorption step would be expected to be slowest, the reaction appears to be zero order, whereas at higher partial pressures of oxygen, first-order behavior is observed (references 4, 5, 8, and 9). No explanation for this appears to have been advanced.

Attempts to determine the primary products of the graphite-oxygen reaction are complicated by secondary reactions between CO and oxygen to yield CO_2 and between graphite and CO_2 to yield CO. The apparent ratio of CO to CO_2 appears to increase markedly with temperature. However, it is debatable whether this results from an increase in the ratio of CO to CO_2 formed by the primary reaction or from increased reaction of CO_2 with graphite.

The influence of reactant diffusion on the rates of the carbon-gas reactions has been discussed at length by Walker et al.,[5] Wicke,[10] Dodson,[11] Robinson,[12] and others. The surface available for reaction, because of the porous nature of the graphite, may be many times greater than the external geometrical surface. Since the pores are quite small, however, the rate of reaction at the internal pore surfaces may be dependent on the rate of diffusion of gases to and from these surfaces. Where the rate of surface chemical reaction is rapid relative to the rate of diffusion, it would be expected that the reactant gas would be entirely consumed before it could penetrate far into the graphite, and thus the reaction would be confined to a narrow zone near the surface of the sample. This type of behavior has been observed at high temperatures, above 500 to 600°C, for the graphite-oxygen reaction. However, if the surface reaction rate is slow relative to the rate of reactant diffusion, the reactant will be distributed uniformly throughout the graphite pores, and the reaction will occur uniformly throughout the bulk of the sample. For the graphite-oxygen reaction, this type of behavior is observed at temperatures below 300 to 400°C.

At temperatures above 800 to 900°C, where the chemical reaction rate is extremely rapid, the over-all reaction rate may become dependent on the rate at which reactant gases are carried to the external surface by diffusion and convection from the bulk gas stream. Under such conditions, temperature variations become less important in determining over-all reaction rates, but the velocity of gas flow past the sample exerts a marked influence since the rate of mass transfer of reactants to the graphite surface is strongly dependent on gas flow velocity.[13,14]

The combined effects of chemical reaction, diffusion in the graphite pores, and diffusion into the bulk gas phase lead to the establishment of various characteristic reaction "regimes" in which different effects predominate. These were recognized by Wicke[10] and have been discussed at length by Walker et al.[5]

Choice of a Reaction-Rate Basis

The combined effects of diffusion and chemical reaction make it difficult to choose a basis for expressing graphite-oxygen reaction rates that will be independent of sample size and geometry. Under certain conditions, diffusion might not limit oxidation rates in a small sample of graphite with a high ratio of surface area to volume and yet might exert a considerable retarding influence in a larger sample of the same graphite under the same conditions. In such cases, attempts to predict large-sample oxidation rates from small-sample experiments could be in serious error if the oxidation rates were expressed on a simple basis such as sample mass, volume, or surface area.

The situation can be improved somewhat by introducing correction factors to account for the effects of diffusion on reaction rates. These factors are based on solution to the differential equations describing simultaneous diffusion and chemical reaction in a porous solid. A brief description of the approach is given in reference 6; more extensive discussions are given by Walker et al.,[5] Dodson,[11] Robinson,[12] and others.

In this correlation the reaction rates are expressed on the basis of the "active mass" of the oxidizing sample. The active mass is defined as some fraction, η, of the total mass, where η is the ratio of the observed reaction

rate to the rate that would be expected if the external gas composition existed uniformly throughout the pores of the sample. The fraction η is estimated from the solutions to the differential equations describing simultaneous chemical reaction and diffusion in a porous solid. Methods of calculating η have been presented by Robinson[12] and the reviewer.[6] It must be emphasized that such calculations are highly approximate, since they cannot account for the effects of secondary reactions within the pores, unequal reactant and product gas volumes, and the changes that occur in pore size and structure as reaction proceeds. However, they do yield at least semiquantitative agreement between predicted and observed effects of diffusion on reaction rates,[12] and they provide the best available method of correcting for these.

Among the factors, other than diffusion, that can exert an important effect on the rates of the carbon-oxygen reaction are the size and orientation of the graphite crystals, the presence of solid-phase impurities, neutron or gamma radiation, and certain gas-phase contaminants. The importance of crystal size and orientation has been extensively investigated and is well summarized by Walker et al.[5] The edges of graphite crystals are about 10 to 200 times more reactive to oxygen than the basal planes under most conditions. Hence, graphite-sample reactivity would be markedly dependent on the relative proportion of crystals with basal planes oriented parallel and perpendicular to the surfaces. Possible variations in orientation throughout a graphite sample could account for some of the reported variations in reactivity from point to point in a sample.[15]

A number of investigators[16–18] have found marked acceleration of the graphite-oxygen reaction by inorganic impurities. Earp and Hill[16] report, however, that phosphates provide a temporary inhibition of reaction. Hennig[19] noted a significant increase in the reaction rate due to small amounts of adsorbed water and proposed a mechanism to account for this. A possible mechanism for inorganic catalysis is briefly discussed by Walker et al.[5]

Many investigators have reported marked acceleration of the graphite-oxygen reaction by both neutron and gamma radiation in the temperature range of 200 to 400°C.[20–27] Kosiba and Dienes[25] explain the effect of neutron irradiation in terms of graphite atom displacement resulting from neutron bombardment leading to a higher concentration of active sites for adsorption than in unirradiated graphite. Davidge et al.[22] and Tomlinson[27] present evidence that gamma radiation increases the oxidation rate through gas-phase production of ozone (O_3), which reacts with graphite more rapidly than molecular oxygen. At temperatures above 500°C, the effect of radiation on graphite oxidation rates (both by oxygen and CO_2) appears to become unimportant.[18,28] This would be expected since, at higher temperatures, neutron-induced radiation damage in the graphite should be rapidly annealed and the rate of recombination of ozone to oxygen should be rapid enough to prevent significant ozone concentrations in the gas phase. An experimental program for further study of radiation effects is presently under way and has been described by Dahl.[21]

The presence of certain halogen-containing gases has been shown to inhibit the carbon-oxygen reaction to a marked degree. Arthur[29] and Wicke[10] have noted a strong retardation in rate produced by phosphoryl chloride ($POCl_3$). Day et al.[13] demonstrated a dramatic rate reduction due to chlorine with an experiment in which a graphite specimen burning at 2000°K in an air blast was extinguished by the addition of only 0.25 per cent Cl_2 to the air. Smaller chlorine concentrations did not quench the reaction completely, but they did lead to a marked retardation that persisted for some time after chlorine addition ceased. Dahl[21] has investigated the effect of chlorine on graphite oxidation in some detail and has advanced a mechanism based on preferential adsorption of chlorine on active sites on the graphite. Quantitative agreement with experimental results is shown.

Correlation of Rate Data

It is desirable to express graphite-oxygen reaction-rate data in terms of a parameter that can be independent of sample size and geometry and oxygen partial pressure. Differences between results of different investigators would then, hopefully, be due only to uncontrolled variations in the graphite used and not a result of changed experimental conditions. In approaching this objective, the reviewer found that the following principles were useful for comparing data:

1. Only results of experiments with high-purity artificial graphite were included.

2. The chemical oxidation rate was assumed to be first order with respect to oxygen partial pressure.

3. Diffusional effects were taken into account by basing all rates on the active mass of graphite, as discussed above.

Graphite oxidation-rate data from eight literature sources were expressed in terms of a reaction-rate constant, k, whose units are

$$\frac{\text{Mass of graphite reacting}}{(\text{Active mass of graphite sample})(\text{hr})(\text{atm of } O_2)}$$

The reaction-rate constant k was calculated from the observed rate constant, K, by the relation

$$k = \frac{K}{\eta} \qquad (1)$$

The observed rate constant is, of course, determined by the volume and external surface area of the oxidizing sample and by the effective oxidation depth. Diffusion coefficients for oxygen within the sample pores, as given by Dodson,[11] were employed. Corrections for other types of graphite and experimental conditions were made by assuming that the diffusion coefficient was proportional to the square of the sample porosity,[5] the $^3/_2$ power of the absolute temperature, and the -1 power of the absolute pres-

Table II-1 GRAPHITE OXIDATION RATES AND EXPERIMENTAL CONDITIONS

Data source	Sample geometry and dimensions	Sample porosity	Measurement temp., °C	Observed reaction-rate constant* (K)	Correction factor (η)	Active mass rate constant† (k)
Earp and Hill[16]	Rectangular block, 5.4 × 0.95 × 0.95 cm	0.24	450	5.32×10^{-4}	1	5.32×10^{-4}
		0.24	500	3.28×10^{-3}	1	3.28×10^{-3}
		0.24	550	1.51×10^{-2}	0.983	1.54×10^{-2}
		0.24	600	7.60×10^{-2}	0.856	8.88×10^{-2}
		0.24	650	2.65×10^{-1}	0.512	5.18×10^{-1}
		0.16	450	2.06×10^{-4}	1	2.06×10^{-4}
		0.16	500	1.15×10^{-3}	0.986	1.16×10^{-3}
		0.16	550	5.75×10^{-3}	0.983	5.85×10^{-3}
		0.16	600	3.45×10^{-2}	0.802	4.31×10^{-2}
		0.16	650	1.22×10^{-1}	0.415	2.94×10^{-1}
		0.16	700	3.32×10^{-1}	0.158	2.10
Dahl[21]	Hollow cylinders, 0.426 in. OD, 0.25 in. ID, 2 in. long	0.227	550	1.81×10^{-2}	1	1.81×10^{-2}
		0.227	600	5.67×10^{-2}	0.974	5.83×10^{-2}
		0.227	700	6.39×10^{-1}	0.918	6.97×10^{-1}
		0.227	750	1.81	0.334	5.43
Gulbransen and Andrew[9]	Thin plates, 4 × 0.78 × 0.05 cm	0.233	475	1.95×10^{-3}	1	1.95×10^{-3}
		0.233	575	3.54×10^{-2}	1	3.54×10^{-2}
Robinson[12]‡	Cylinders, 6 in. OD, no length given	§	485			2.39×10^{-3}
			500			5.76×10^{-3}
	Rectangular blocks, 2 × $^1/_2$ × $^1/_2$ in.		450			5.72×10^{-4}
			500			2.43×10^{-3}
			550			8.10×10^{-3}
Kosiba and Dienes[26]	Solid cylinder, $^1/_8$ in. OD, 2 in. long	0.27¶	350	2.98×10^{-6}	1	2.98×10^{-6}
		0.27	400	6.14×10^{-5}	1	6.14×10^{-5}
		0.27	450	1.10×10^{-3}	1	1.10×10^{-3}
		0.27	350	1.58×10^{-6}	1	1.58×10^{-6}
		0.27	400	2.76×10^{-5}	1	2.76×10^{-5}
		0.27	450	3.59×10^{-4}	1	3.59×10^{-4}
Kosiba and Dienes[25]	Solid cylinder, $^1/_8$ in. OD, 2 in. long	0.27	300	1.17×10^{-6}	1	1.17×10^{-6}
		0.27	350	1.08×10^{-5}	1	1.08×10^{-5}
		0.27	400	1.45×10^{-4}	1	1.45×10^{-4}
		0.27	450	8.15×10^{-4}	1	8.15×10^{-4}
Currie et al.[31]	Solid cylinder, 1.25 in. OD, 2 in. long	0.25	600	1.14×10^{-2}	0.852	1.34×10^{-2}
		0.25	700	1.58×10^{-1}	0.138	1.14

*In units of mass of graphite reacting per mass of sample per hour per atmospheres of O_2.
†In units of mass of graphite reacting per "active" mass of sample per hour per atmospheres of O_2.
‡Reported values had already been corrected to active mass basis.
§Reference 12 does not give sample porosity values.
¶Remaining values in this column are assumed values.

sure.[30] Over-all observed reaction rates and the calculated values of k are given in Table II-1 (references 9, 12, 16, 21, 25, 26, and 31). Methods of data reduction and a sample calculation are presented in reference 6.

The rate constant k is plotted in Fig. II-1 (based on data from references 9, 12, 16, 21, 26, and 31) as a function of the reciprocal of the absolute temperature. The results are represented by the equation

$$k = 7.24 \times 10^9 \exp\left(\frac{-22{,}000}{T}\right) \qquad (2)$$

where T is in degrees Kelvin.

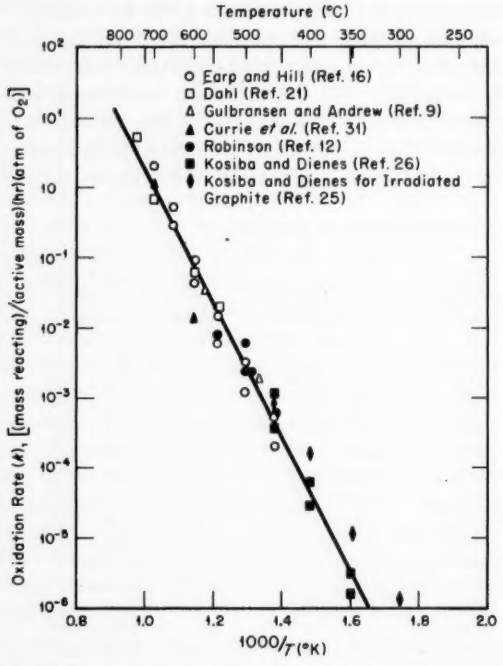

Fig. II-1 Graphite oxidation rate as a function of temperature.

The activation energy obtained from the above expression is 44.0 kcal/g-atom. This is within the range of values reported from the individual investigations[5] but is somewhat below the average (about 50 kcal/g-atom). A few points representative of irradiated graphite[25] have been included in the plot, although they were not used in determining Eq. 2. Below 400°C these points lie above the line representing Eq. 2 by a factor of 5 or more, but they appear to approach the line rapidly near 500°C. Points for unirradiated, high-purity graphite agree within a factor of 6 or less. It is felt that the major cause for such discrepancy lies in variations among the graphite specimens investigated with respect to impurity content, crystal orientation, pore-size distribution, etc.

Use of Rate Data in Reactor Hazards Calculations

Even after suitable corrections have been applied, there is still a considerable spread in reported graphite oxidation rates. This is due in part to the necessarily approximate nature of the corrections and in part to the uncontrolled factors affecting graphite reactivity. DeHalas and Dahl[32] have suggested the use of an empirical "reactivity parameter" for a given graphite to account for the latter effects. However, computation of the reactivity parameter for a given graphite would require actual oxidation-rate data from representative samples of the graphite. Such data could, of course, be used directly in a hazard calculation.

The question of primary importance in a graphite combustion-hazard calculation is the definition of the heat-release rates as functions of time and position. To define these rates accurately would require detailed knowledge of the gas composition and temperature profiles within the graphite pore structure, as well as the rates of the graphite-oxygen, CO-O_2, and graphite-CO_2 reactions as accurate functions of temperature and composition. It is probably wiser at this time to calculate the heat-release rate as the product of an over-all oxidation rate and an effective heat of reaction. In most reactor hazard calculations, oxygen is considered to be present in considerable excess,[1] and the effective heat of reaction would probably be close to that for complete oxidation of carbon to CO_2 ($\Delta H \cong -94{,}000$ cal/g-atom of carbon). So long as oxygen is not strongly depleted in the reactant gas mixture, any secondary CO-O_2 reaction will probably occur primarily within the graphite pores or in a narrow zone close to the graphite surface. Hence heat release for complete oxidation would occur within, or close to, the graphite itself.

The spatial distribution of heat release within a sample of oxidizing graphite will depend on the relative rates of oxygen diffusion and chemical reaction in the sample pores. For conditions where diffusion is rapid relative to reaction,

the oxygen concentration, reaction rate, and heat-release rate will be essentially uniform throughout the reacting sample. Where reaction is rapid relative to diffusion, these quantities will decrease approximately exponentially from maxima at the outer surface to zero at some distance from the surface. The over-all effect of the latter condition is the same as if the reaction occurred at a uniform rate for a distance L from the outer surface and were zero throughout the remainder of the sample. This "effective oxidation depth," L, can be estimated from the relation

$$L = \left(\frac{MD_p}{k\rho RT}\right)^{1/2} \qquad (3)$$

where

M = molecular weight of carbon, 12.01
ρ = density of graphite, mass/volume
D_p = effective diffusivity for oxygen in graphite pores, (length)^2/time
R = gas constant
T = absolute temperature

Values obtained from Eq. 3 have been plotted as a function of temperature in Fig. II-2 for graphite with typical properties. When reaction and diffusion rates are of comparable magnitudes, the heat-release rate will still decrease from the surface toward the center of the sample, but it will not vanish.

Conclusions

Oxidation-rate data for high-purity artificial graphite can be correlated within a factor of 6 by Eq. 2. Closer agreement is precluded by the approximate nature of the methods used to correct for the effects of reactant diffusion in the graphite pores and by the variations in such factors as crystal size, orientation, impurity content, and pore structure from one graphite to another.

Heat-release rates for reactor hazard calculations may be estimated on the assumption that graphite is oxidized completely to CO_2 and that all heat from the reaction is released uniformly in a region extending from the graphite surface inward a distance L from the surface, where L is given by Eq. 3. It is felt that the uncertainties inherent in heat-release-rate calculations do not justify more refined methods at present. (J. W. Prados)

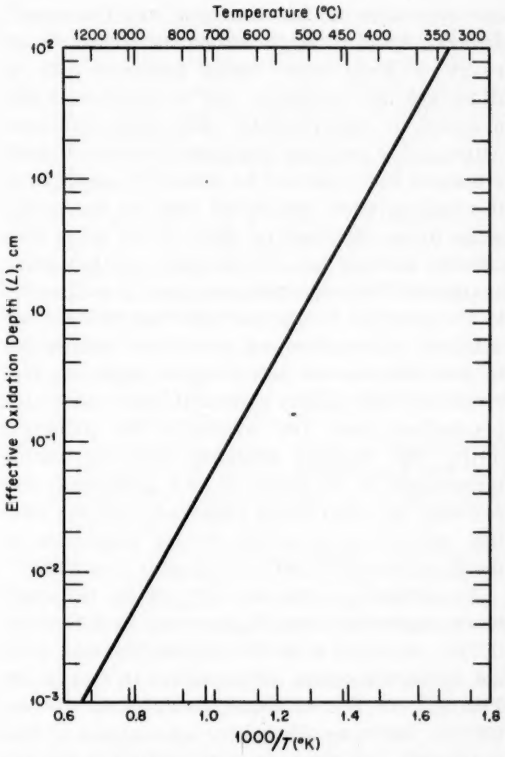

Fig. II-2 Effective oxidation depth L vs. temperature (for $\epsilon = 0.24$).

Application of Statistical Analysis to the Hot-Spot Problem

The cumulative effects of perversities in flow distribution superimposed on variations in power distribution and fuel loading may lead to local fuel-element temperatures much above the nominal temperatures indicated by the average values of the heat-transfer coefficient, fluid flow, and power density. The problems are inherently extremely complex, since they depend on such factors as irregularities in dimensional tolerances in production fuel elements and vagaries in fluid-flow distribution. The three papers reviewed here are concerned with the application of statistical methods as a means for appraising the probability of the various hot-spot factors having cumulative effects in a reactor.

The first paper, "The Application of Statistical Methods of Analysis in Predicting Burnout Heat Flux,"[33] applies statistical methods to

the approach of Le Tourneau and Grimble.[34] Several specific applications to water-cooled reactors were made using assumed data to illustrate the technique, but no production tolerances or experimental data were included. Although the analysis has been developed in such a manner that it should be generally applicable, it requires good statistical data on the variations to be expected in each of the many hot-channel factors for the specific reactor being analyzed. The technique presented is well suited to the analysis of reactor types for which much detailed information on production tolerances is available and for new designs employing fuel elements that closely resemble some for which production data are available. Its principal utility for designs entailing new fabrication techniques is to point up the problems, especially the tolerances required, and the data that should be recorded during inspection of the first core both before and after operation.

An estimate of the severity of the hot spots to be expected in the Engineering Test Reactor (ETR), together with the probability that such hot spots will occur, is presented in a paper by Tingey.[35] In lieu of experimental data for the ETR to serve as a basis for an analysis of this character, the limited data available for the Materials Testing Reactor (MTR) were employed. Information on variations in fluid flow from one production fuel element to another was included, but the local velocity distribution within a given fuel-element assembly was not included. No statistical data on fuel-element dimensional tolerances were given nor were any data supplied on changes in fuel-element dimensions which might occur during the course of operation because of warping or other factors. As is ordinarily the case in hot-spot studies, the investigator was forced to employ many assumptions to fill in gaps where information was not available. The report was prepared primarily for internal consumption, and hence it presents very specific calculations that are difficult to follow for one not intimately acquainted with the ETR.

The third report reviewed, "Prediction of Fuel-Element Failures During Accident Conditions,"[36] was prepared as a portion of the hazards analysis of a preliminary design study for the Experimental Gas-Cooled Reactor (EGCR). This paper is concerned with the application of the results of an earlier hot-spot analysis to the estimation of the probability of failure of fuel-element capsules during the transient conditions following either a blower power failure or a rupture in the main pressure envelope which could cause the system gas pressure to drop to atmospheric. The techniques and data presented in this paper have been made obsolete by later estimates,[37] but the approach employed may be of interest to others confronted with similar problems in the design of gas-cooled reactors employing UO_2 encapsulated in stainless steel. (A. P. Fraas)

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Section III

INSTRUMENTATION AND CONTROLS

Redundancy and Coincidence in Reactor Safety Systems

In an effort to provide adequate safety, to assure a high degree of reliability, and to prevent accidental reactor shutdown, the use of redundancy and coincidence in safety systems is being widely advocated. Several articles in the current literature are devoted to studies of the relative merits of various coincidence techniques. A popular approach has been described by Jacobs[1] in a report that is the basis for the following discussion.

Starting with a number of precise definitions and assumptions, Jacobs proceeds to make a comparison, on a mathematical basis, of a number of different series-parallel arrangements of contacts. By postulating certain types of failures, he assesses each arrangement both as to its safety and its rate of false shutdown. The orderly presentation and the results obtained lead to the conclusion that much is to be gained in operating continuity, with negligible loss in safety, by using a coincidence system for reactor safety instrumentation channels.

Jacobs points out, however, that the assumptions leading up to the calculations should be considered seriously because they are based on random failures and probabilities. In any complex control and safety system with a number of similar channels of instrumentation, the designer must be extremely careful if he is to maintain independence between like channels. In fact, independence alone may not be enough. It is quite probable that channels used in coincidence should be basically different to prevent some design or maintenance error from being common in all channels, as has been proposed by Siddall.[2] One of the necessary conditions for the attainment of performance figures such as those given by Jacobs is the attainment of an extremely high degree of independence among instrumentation channels.

Another feature that appears in the calculations is frequency of testing. Inasmuch as Jacobs assumes that some test is made, it is presumed that a test mode is available which makes possible the detection of failures that could contribute to a false shutdown or could leave the reactor unprotected. This presumption must be examined in detail. Tests must not shut down the reactor, and yet they must prove that the reactor could be shut down by a particular device. The tests must not, while being carried out, reduce the effectiveness of the safety system. To be adequate, the test must start at the sensor and be carried through to a point as close to the shutdown mechanism as is possible. Jacobs states that "it is important that the interval between tests be short if safety is to remain high." It is obvious that any inadequate test implies the equivalent of a long interval between tests, and hence less safety. The point to be considered is not whether some test is performed once a shift or once an hour but whether a valid test capable of detecting a failure anywhere in the system is performed. Since the test plays such a vital part in maintaining good safety system performance, the test and the person or device performing the test are both integral parts of the system. They must not be taken for granted.

A third feature which is included in the calculations and which seems easy to understand is repair. With the discovery of a failure comes the necessity for repair or replacement. In many coincidence systems, devices of one sort or another are used to bypass failed components. Inherent in most of these bypass devices are modes of failure which can either reduce safety or allow unnecessary shutdowns to take

place. Such devices should be carefully designed and subjected to occasional testing.

An aspect of the use of coincidence to provide longer continuous operating cycles, which was not mentioned in Jacobs' article, has come to the reviewer's attention in recent months. Although most articles on coincidence make specific reference to safety systems, it has been found that once a design is committed to coincidence there are few places in the control system which are not influenced by this choice. Although safety may not be involved, there are many places in a reactor control system where failure of a single device can shut down the reactor. Therefore, if one decides to use coincidence in safety, he must carry this decision deep into the control system or he will not get the full benefit he expects.

In pointing out difficulties and limitations in the application of coincidence techniques, there is no intent to criticize the excellent work done by Jacobs and others in the use of mathematical methods in the evaluation of various coincidence arrangements. Rather, an attempt has been made to show that the satisfying of the assumptions on which the calculations are based is an almost impossible task in a complete system. If, then, a system does not satisfy all the conditions in the mathematical model, great care must be used in assessing the validity of the conclusions drawn from the calculations. The presence of a single "Achilles' heel" in a system can go far to negate the benefits of a well-thought-out coincidence arrangement. The weak points in a system may be in inadequate tests, control-system weaknesses where coincidence appears to be unnecessary or too difficult to attain, faulty designs or faulty maintenance procedures which make all channels of a type fail at the same time and for the same reason, or other less-obvious shortcomings. Only a detailed study of the entire system can provide information necessary to evaluate the over-all safety and reliability of a system. It is quite possible that the results of such an evaluation will show little resemblance to the results obtained by a mathematical analysis of the coincidence technique used. (S. J. Ditto)

Control Rods and Control-Rod Drives in Power Reactors

One of the most important requirements of any reactor safety system is the ability to shut down the reactor in an emergency and to hold it positively shut down. The degree of safety which is achieved in this operation is determined principally by the reactor control rods and their drive mechanisms, although any temperature effect on reactivity may affect the transient behavior of the reactor during the shutting-down process, as well as the final shutdown requirements for the control rods. This study is concerned principally with the shutdown features of control rods and their drive mechanisms, rather than with the aspects of reactor control during normal power operation.

The importance of the rods and their drives to reactor safety has been recognized since the inception of nuclear technology, but, as reactors become more and more numerous and as design objectives become more complex and difficult to achieve, there may be a tendency to lose sight of some of the basic criteria for the design of such systems. Concern over the degree of compliance with some basic safety criteria was expressed by the ACRS in a recent letter to AEC Commissioner John F. Floberg. The following statements[3] are excerpts from that letter.

> The Committee wishes to point out that the present state of the art of reactor control rod design leaves much to be desired, in particular for application to power reactors where reactivity is increased if the rod falls downward under the influence of gravity. It is important to stress the simplicity, ruggedness and reliability of control rod drives and to insure that the position of a rod is correctly indicated even should the rod be separated from its drive. Design problems appear to be particularly difficult for hydraulic rod drives. A control-rod-fall accident should be made as nearly impossible as practical.
>
> We believe that this recommendation should be transmitted to all reactor designers concerned.

In this review an attempt is made to enumerate some pertinent design criteria and to evaluate them from the point of view of reactor safety. This discussion is followed by a review of the control-rod-drive mechanisms of a number of power reactors now operating or proposed for construction. The reactors whose rods and drives are discussed in this article were somewhat arbitrarily selected from among those power reactors on which some information has been published, although in several instances the data are preliminary. Furthermore, the extent to which the rod-drive systems of

these reactors have been reviewed has only been with regard to the attainment, or lack thereof, of certain inherent characteristics that may be recognized as being desirable in such systems. This review does not purport to establish the adequacy or safety of any particular control-rod-drive system, since those systems which are inherently preferable may be employed so as to negate their advantages and those which have inherent disadvantages may be employed in such a manner that the disadvantages may be overcome. It would be fair to conclude, however, that the necessary safety functions of the control rod and its drive system may be more readily and reliably attained if the system possesses the desirable characteristics described below than if these characteristics are lacking. The imposition of additional instrumentation, mechanisms, and operations or control circuitry does not, in general, constitute an acceptable alternate for proper design, since the auxiliary devices that can be used have, on the average, far less reliability than can be tolerated for such service.

General Criteria

Some general criteria are discussed below which are believed to apply to any reactor that requires control rods or analogous devices to shut it down or hold it down at any time. The examinations of the designs and installations of reactor control-rod-drive mechanisms undertaken here are made in the light of principles that have been developed over the years by ORNL personnel for dealing with the safety aspects of the nuclear reactors designed at the Laboratory. Although the application of these principles has, in the past, been mainly to the light-water-moderated research type reactor using highly enriched fuel, it appears that these principles may be equally valuable with respect to many power reactors if they are carefully considered. Unless otherwise indicated, the principles and discussions are based on the use of absorber type control rods. However, these methods can be applied to fuel type control rods by taking the rod positions and direction of motion with respect to the core as opposite. For convenience the principles are stated herewith.

1. The first requirement relates to the minimum acceptable total worth of the rods. It is important that the minimum worth be sufficient so that no single error in manipulation of rods or fuel in the reactor can cause it to go critical inadvertently. This is particularly important under shutdown conditions since the vessel cover may be removed, and thus operating procedures and protection might not be effective. The required shutdown margin is determined both by the maximum expected loaded excess reactivity and by the worth of the strongest single control rod. At ORNL the objective for research reactors is to provide a 2:1 ratio of the total rod worth to excess reactor loading, based on normal loading and a cold, clean core. As discussed below, there may exist mitigating circumstances in power reactors which permit the use of a lower ratio. In any event, sufficient margin would be required so that any one rod may be completely withdrawn without any possibility of the reactor becoming critical.

2. The controllable reactivity must, for safety reasons, be divided among several independently driven control rods. In selecting the worth and therefore the number of rods, the division must be such that the simultaneous loss of a significant number of the most valuable rods will not prevent a scram and the subsequent holddown of the reactor whenever necessary and for as long as is required. Power reactors often require a larger multiplicity of control rods because they often contain more than one critical mass, each of which must be positively controlled.

3. The control rods must never be subjected to an unbalanced force resulting from equipment failure, misoperation, or any other credible accident situation that would result in their being inadvertently withdrawn or ejected from the core.

4. No ultimate safety action designed to protect a reactor should ever be prevented by any failure or credible combination of failures or misoperations of the rod-drive mechanisms or their auxiliary systems, when such action is demanded. In the case of research reactors, the ultimate safety action is a scram, but in power reactors it may take other forms. In all cases, however, the point is that the safety action must be immediately available on demand under all circumstances.

5. In stationary reactors, control rods should not be required to remain in their seats against the force of gravity, nor should scrams be effected against gravity. Any necessary initial acceleration above that of gravity may be obtained by mechanical or other suitable means,

the failure of which should not interfere in any way with the normal rod acceleration by gravity.

6. There must be a positive bottom or a fixed and definite seated position for the control rods in the core. No failure in the rods or rod-drive mechanism should ever permit any rod or rods to fall through or out of the core. Experience indicates that a stop on the drive mechanism is not a substitute for a bottom in the core.

7. The position of all rods must be known or be quickly ascertainable at all times. Of particular importance is the knowledge that the rod is in contact with its drive and/or is seated.

8. Under no circumstances should the rod drives be capable of withdrawing control rods at rates in excess of those established in the reactor design as being safe. In some types of reactors, the safe rate is very high, and the addition could even be in the form of a step increase.

9. The control rods must never be able, in their movement, accidentally or unintentionally, to displace in any direction any fuel considered to be part of the reactor core or blanket. This, of course, does not apply to fuel that may form part of a control-rod assembly.

Some of the reasoning and experience behind these principles and the ramifications of their application may be enlightening. There are, for example, some very practical reasons for the use of many rods, and at least one reason is vital from the safety standpoint. No single rod or rod-drive mechanism can be relied upon to function continuously without failure. With a sufficiently large number of rods, individually and independently driven, the loss or malfunctioning of a reasonable number of them will not jeopardize the safety of the reactor in scram or shutdown situations. On the other hand, there can be danger in numbers since rod failures can induce accidents, especially in cases in which complicated mechanisms are used. This is true not only because of the increased probability of failures or malfunctioning but also because of "made" problems not basic to a particular design or concept. These problems are sometimes occasioned by the inevitable divided responsibility for maintenance of the equipment, and they are surprisingly frequent and difficult to avoid. One good reason for providing considerable excess worth in the control rods is to allow more margin for rod and drive failure before the protection afforded the reactor is seriously reduced.

No criteria for determining the shutdown margin seem to have been established by designers for application to power reactors. Unlike most research reactors, power reactors often contain several critical masses, each of which must be controllable, and this makes it necessary to distribute a relatively large number of control rods throughout the core. Also, the absorber rods may not be much blacker than the fuel when its enrichment is low. Since each rod is worth very little, no mechanism apparently exists for inserting large amounts of reactivity with rods in a single operation or accident. In power reactors, as in research reactors, there is always considerable incentive (because of lack of space, for example, or considerations of neutron economy) to reduce the number of rods to a minimum. The net effect may be that a relatively small shutdown margin remains, although this is not necessarily in itself of great concern.

What is of concern is that the existence of such a low shutdown margin may indirectly contribute to the occurrence of accidents. Consider the power reactor in which the withdrawal of rods is controlled by a programmer. The withdrawal speed of the individual rods is maximized with consideration of such factors as the number that will be withdrawn simultaneously, their worth, and the rate of activity addition permitted by the thermal design of the reactor. Since the programmer is provided with means for selecting various desired and safe combinations of rods for withdrawal, it is suspected also of being capable of selecting other unsafe combinations in the event of a failure of its internal mechanical or electrical systems. The addition of interlocking and monitoring systems intended to prevent such failures from precipitating accidents does not constitute a solution to the problem at hand. Such systems fall in the category of patches which give a false sense of security. They may fail also, and their failures will most likely not become apparent until they are finally called upon to perform their intended function. Since new and unexpected ways of inducing accidents related to control-rod movements are continuing to be discovered, it seems only prudent to avoid all known ways and to provide a reasonably generous shutdown margin to assist in handling the undiscovered one.

It is of concern that any reactor design should require a means, in addition to the control rods, of providing some excess for holddown, let alone requiring this means for achieving shutdown at all. Since safety is afforded in such cases only when the auxiliary means functions properly, the auxiliary means cannot be permitted to fail. This is a difficult requirement to meet. The present principal application of such extra shutdown means is the addition of soluble poisons to the primary cooling system in water-cooled reactors. It is reasonable to expect that systems using liquid control means should be designed to stringent criteria comparable to those for the more usual solid rods. It is true that, under certain circumstances, there is no need to add the poison in milliseconds or even seconds to effect a scram, and checks and corrections can be made as required. Every reader can visualize ways, however, in which the poison could become so dilute while the cover is off the reactor for refueling that holddown is no longer effected.

The third principle is, of course, self-evident; accidental ejection of the rods must be prevented. Any rod withdrawal tends to put the reactor on a positive period, and therefore may actually produce a hazardous condition. The principle is the same whether the rod is "washed" out by coolant flow, is lifted out when the vessel head is removed for refueling, or is ejected when the counterbalancing force is lost for rods having extensions through a pressure vessel. In reactor design it is much better to avoid the possibility of such situations than to provide one or more backup devices which are, of course, fallible and which may or may not act properly and promptly when needed. Undoubtedly, situations will arise in which use must be made of backup devices. The design solution in these cases must include the provision of a practical means for testing the condition and effectiveness of the protection at frequent intervals without interfering with reactor operation.

With respect to achieving positive safety action on demand, if a reactor depended on the force of gravity acting through a single weight-pressurized accumulator to scram all its rods, obviously a failure in the hydraulic system could easily and directly render the entire system inoperable; such a design is clearly unacceptable. On the other hand, if individual accumulators were provided but their pressure piping was placed so that some single accident could rupture all or a large percentage of it, the result would be the same as before. Without postulating the means of producing an accident responsible for a failure, it is evident that rods which are scrammed or driven by mechanisms working through vessel walls are more vulnerable than those which can be scrammed independently and without interference from the drive or its connecting mechanism.

For holddown of rods, the only method that the reviewer's experience has indicated requires no testing is one which makes use of the force of gravity acting directly on the individual rods to keep them seated. The problem is to make all other forces that may act to unseat the rod, such as those produced by coolant flow, smaller than the direct force of gravity tending to keep it seated. Fixed reactors have an advantage over mobile reactors in this respect, but this advantage is not always realized in their rod-drive design. Ship and submarine reactors simply cannot depend on gravity to drive the rods in. In this case, recourse must be taken to incorporate some built-in means of furnishing the necessary rod driving power, such as a compressed spring. These devices, quite clearly, should be arranged so that they may be tested as frequently as is deemed necessary without interfering with the operation of the reactor.

The need for a "bottom" in a reactor, as far as the control rods are concerned, also seems to be self-evident. The intent is, of course, to guarantee that when the control rods scram they move to a fixed and definite location with respect to the core of the reactor where they can exercise their full worth. The bottom may be inadvertently omitted in many ways. As an example, in the preliminary design of an experimental reactor some years ago, the absorber rods were required to be flexible. One way of providing the necessary flexibility was to arrange the absorber in the form of small cylinders slipped over a central wire support. The whole assembly was suspended from above and arranged to move up and down in a tube through which cooling gas flowed. Not until the experimental assembly was actually under construction was it realized that the cooling-gas tube extended below the reactor core far enough before it changed direction to permit the absorber to fall completely through the core if the support wire failed. The actual loss of a rod in such an

instance may appear, at first, to be only a nuisance, since its effects would be controllable by moving the remaining rods. The real difficulty becomes apparent, however, when it is observed that the lost rod may have acquired sufficient velocity by the time it is leaving the core to add reactivity at a rate in excess of that which the remaining rods are capable of handling.

More recently an incident actually occurred in a power reactor which illustrates the validity of the principle. In this reactor the absorber rods are withdrawn downward out of the core by means of a mechanism capable of exerting substantial forces. One of the rods became disconnected from its operating mechanism and failed to follow when withdrawal was attempted. Later, however, while the mechanism was still withdrawn, the rod for some reason freed itself and fell a short distance; in doing so, it effected a step addition of reactivity. Fortunately the result this time was not serious, but it brought up the question of the wisdom of knowingly building into a reactor the potentialities for generating incidents.

Since the operator cannot see into the reactor, he must depend on instrumentation to give him the information he needs concerning rod position, and there are safety as well as control aspects to the problem. Following a scram demand and during shutdown periods, there should be positive information available, wherever needed, that control rods which are released from their drives for scram are or are not seated or are or are not in contact with their drive mechanisms. Rods that are neither seated nor in contact with their drives are probably jammed. The use of seat, rod-drive contact, and continuous-position indicators is necessary to provide reliability. With this arrangement, the loss of any one sensor does not create a situation of doubt, since the operator can still determine the position of any rod, either immediately or without serious delay, from the other two sensors.

Rod-drive speeds must have a fixed, maximum value that is based on the greatest acceptable rate of reactivity addition. The intent here is to limit the temperature excursions that might result from inadvertent misoperation of the rod drives, and thus avoid the chance of damage to fuel elements, rods, and even core structural members and the like. It is possible to design a reactor in which the inherent temperature coefficients of reactivity prevent the attainment of unsafe temperatures, even if the control rods are moved instantaneously to their fully withdrawn positions. This is the case in the N.S. *Savannah*, for example, and the final position of the rods is the factor that determines the severity of the rod withdrawal accident, not the rate at which the rods are withdrawn to the selected position. In such cases, any rod withdrawal speed acceptable for other reasons, such as operating convenience, may be safely employed. Unfortunately, many reactors cannot be designed in this way, at least in the present state of the art, and the necessary protection must be afforded by limiting the rate of reactivity addition.

Alternating-current (a-c) induction motors powered from a fixed-frequency source, such as a public utility, inherently provide this desired upper speed limit. If the power is locally generated, the frequency must be controlled by a high-grade regulator that is incapable of appreciable upward frequency deviation in case of failure. The choice of induction motors is limited to those which are capable of being instantly and positively reversed, regardless of their speed or direction of rotation, at the moment reversal is demanded. Reversals may be needed to control a number of conditions, a typical one being encountered during startups when short periods can be induced in the reactor by a disorderly withdrawal of the control rods. Any type of induction motor that employs internal inertia switching to disconnect its starting winding is therefore unacceptable. The three-phase induction motor is not a particularly desirable choice either, even though it may appear to be on first thought. Two possible conditions must be considered which may arise at the instant when the motor connections have been switched from the "withdraw" to the "insert" arrangement: (1) a fuse may blow or (2) one of the three contacts of the motor insert relay or starter may not make up electrically. The net effect is to take the rod-drive motor off three-phase power and to reconnect it to a single-phase source. Remembering that a three-phase induction motor runs very well on single phase once it is started, the result is clear; the rod not only does not insert as was demanded, but, what is worse, it continues to withdraw. Unless the fuse that blew was protecting all the drive motors, the situation is probably within the range of control of the unaffected rods;

also, in any case, no intolerable nuclear excursion should result since the maximum rod speed is predicated on uncontrolled continuous withdrawal. The affected rod drive may destroy itself, however, since now the switches that normally furnish overtravel protection are helpless. In other words, since the rod is being withdrawn while the motor is connected to its power source through the insert contacts, the withdraw limit switch, even though properly actuated, cannot stop the rod motion.

The preferred motor is one operated by single-phase power, with identical starting and running windings and the necessary rotating field being developed by a permanently connected capacitor. Reversal is accomplished by switching a feeder wire from one side of the capacitor to the other without disturbing the connection between the motor and the other feeder.

Pneumatic and hydraulic rod-drive systems presently employed are not all inherently free from overspeed problems. In fact, it may be quite difficult to design into the pneumatic system, especially, the necessary positive and accurate speed-limiting features. One possible hydraulic system might be powered by positive-displacement pumps sized to limit maximum rod speeds to safe amounts even under full-flow conditions. Here the burden is placed on the individual pump driving means, and, if a-c motors are used, then the supply-line frequency is ultimately the determining factor.

One of the most effective means for inducing a positive power excursion in a reactor is to bring it to criticality and then drop or shoot some fuel into a void in or near the center of its core. Under controlled conditions and in reactors designed for the purpose, such excursions are useful for testing and evaluating safety systems. Similar excursions may, however, be inadvertently induced in an operating reactor if a control rod, by sticking or other means, manages to lift a fuel element during withdrawal and then suddenly releases it to fall back into the core. In some types of reactor, this possibility is minimized by mounting and supporting the rod-drive mechanisms on special fuel elements so that, when the rod is withdrawn, the force acting upward on the rod reacts downward on the fuel element to keep it in place. The rod is contained within the special fuel-element assembly and cannot contact any adjacent elements within its range of motion. When it is necessary to mount the drive mechanism in other ways, contact between the fuel elements or assemblies and the control rods must be prevented. One successful means of preventing such contact is to use guide tubes for enclosing the rods. These may be anchored directly to the core grid plate (not the fuel elements), and thus to the reactor superstructure; or they may be anchored by various other means, so long as no force is transmitted from the rod or its drive mechanism which would tend to displace any fuel element.

Although the following two factors concerning the use of control rods are not a part of the rod or drive mechanism design, they do bear on reactor safety: (1) seated rods are powerless to aid in effecting a scram and (2) fully withdrawn rods are usually not immediately effective in a scram situation. Since rod positions have a bearing on their shutdown capabilities, it is customary in research reactors to withdraw all rods simultaneously, as far as conditions will permit, while the reactor is in operation. The ultimate safety action usually selected for protection of power reactors, as well as research reactors, is the scramming of the control rods. If, however, the startup procedure requires that rods be withdrawn individually, there are times when some are seated and some are fully withdrawn. Especially in reactors that can be made critical with only a small percentage of their rods withdrawn, scram protection may be somewhat limited because of the rod positions during startup. This is, of course, just the time during which difficulties are to be expected; and, in reactors that cannot safely depend upon temperature or other self-limiting means, it is vital that the rods be arranged so that it is possible, by scramming, to limit the excursion to an acceptable maximum and to shut the reactor down as desired.

Power reactors cannot afford the luxury of the large factor by which the burnout level exceeds the operating level, as ordinarily found in research reactors. For various valid reasons, the flux patterns in power reactors must be carefully adjusted and maintained by manipulation of individual control rods while operating in or near the "power" range. It is not clear, however, that, during startups when flux distribution is not of paramount concern, programmed withdrawal of rods, either singly or by groups, can be justified from the safety standpoint. Startups effected by withdrawing all scrammable rods simultaneously at the most

rapid rate permitted by the reactor design until criticality or, possibly, the operating temperature is reached will leave all these rods in their best possible condition to control any incidents that might occur. Thereafter the rods may be individually adjusted to achieve the desired flux patterns according to a predetermined schedule. Programmed withdrawal procedures are suspect from standpoints other than their limited capabilities with respect to startup protection, as noted earlier.

Boiling-water reactors have their special problems, as do most other types. Rods that are drawn out the top of the core are relatively ineffective when moving through the boiling zone because of the effect of the steam voids. When the rods are inserted to reduce the power being developed in the core, the volume occupied by the steam is reduced, thus increasing the apparent excess reactivity in the core. The net effect is that the control rods must move a considerable distance in scram situations to drive the reactor subcritical. For this reason, designers sometimes arrange the rods to move in and out of the bottom of the reactor in order to keep them in the nonboiling section of the core, and therefore in a more effective portion of their range. Unfortunately in this case, scramming must be done against gravity, and, almost invariably, there turns out to be no bottom in the core when absorber type rods are selected. Reactors may be controlled effectively by pulling or pushing fuel up into the core, as well as by removing absorbers. Undoubtedly there are problems to be solved in following this approach, but it will permit application of the principles in question.

In summary, it is clear that the principles stated above are basic and therefore applicable to power type nuclear reactors. The fact that certain problems in reactor design may be minimized or circumvented by violating one or more of these principles does not invalidate them. Instead, it would seem that insufficient development has been undertaken to permit the completion of a suitable design. Therefore, for safety, it may be concluded that:

1. The total worth, number, and arrangement of the control rods must be such that no single error in manipulation of fuel or rods or the loss of a significant number of the most valuable rods can prevent shutdown of the reactor when required or holddown thereafter for as long as is necessary.

2. It must be expected that credible combinations of accident situations or misoperation or failure of equipment will not at any time result in the addition of reactivity, either inadvertently or at an excessive rate; prevent the ultimate safety action; or permit the motion of the rods to effect displacement of any fuel which, by design, was intended to be stationary with respect to the remainder of the core.

3. The shutdown position of the control rods must be fixed and determined independently of motion or failure of the rod drives themselves.

4. Information relative to the position of all control rods must be available to the reactor operator at all times; of particular interest is knowledge that the rods are seated or in contact with their drives.

Design Descriptions

The information included in this review was collected principally from hazards summary reports.[4–22] In some cases, when the data in the references were in doubt or out-of-date because of design changes, the most recent unpublished data were made available by the responsible parties, as indicated. The information presented should therefore represent the present design criteria or be descriptive of the "as-built" equipment in most cases. Some of the information given below is repeated in Table III-1, along with other data pertinent to the reactors and their control systems. It is interesting to note that, in most cases, the scram time was given as that required to produce complete, or a large percentage of complete, insertion. A frequently used and much more meaningful statement includes information as to how quickly the rods can be inserted sufficiently to turn back any excursion that is controllable by the safety system.

Industry has provided a number of different designs of control-rod-drive mechanisms for incorporation in power reactors. Generally speaking, these may be separated into several rather distinct types, for example, the magnetic jack, rack and pinion, nut and lead screw, cable and winding drum, hydraulic and pneumatic cylinders, and, finally, combinations of these.

Magnetic-Jack Drive Mechanism. The Yankee,[4] Piqua,[5] and Saxton[6] reactors each employ the magnetic-jack drive mechanism. Since this is an unusual mechanism, it will be described before it is discussed as a control-rod drive.

(Text continues on page 29.)

Table III-1 CONTROL RODS AND CONTROL ROD DRIVES IN VARIOUS NUCLEAR POWER PLANTS

Full Name of Plant	Short Name of Plant	Reactor Type	Location	Status as of 9-1-60	Docket No.
Shippingport Atomic Power Station	PWR	Pressurized water	Shippingport, Pa.	Government-owned reactor, operation approved	
Consolidated Edison Thorium Reactor	Indian Point	Pressurized water	Buchanan, N. Y.	Licensed reactor, construction permit issued	50-3
Yankee Nuclear Power Station	Yankee	Pressurized water	Rowe, Mass.	Licensed reactor, interim operating license issued	50-29
Nuclear Merchant Ship Reactor	N.S. Savannah	Pressurized water	Camden, N. Y. (startup)	Government-owned reactor, operating approval pending	
Saxton Reactor Facility	Saxton	Pressurized water	Saxton, Pa.	Licensed reactor, construction permit issued	50-146
Vallecitos Boiling-Water Reactor	VBWR	Boiling water	Pleasanton, Cal.	Licensed reactor, operating license issued	50-18
Dresden Nuclear Power Station	Dresden	Boiling water	Morris, Ill.	Licensed reactor, operating license issued	50-10
Elk River Facility	Elk River	Boiling water	Elk River, Minn.	Government-owned reactor, construction permit granted	PP-1
Pathfinder Atomic Power Plant	Pathfinder	Boiling water with superheat	Sioux Falls, S. D.	Licensed reactor, construction permit issued	50-130
Humboldt Bay Power Plant	Humboldt Bay	Boiling water	Eureka, Cal.	Licensed reactor, construction permit issued	50-135
Boiling Nuclear Superheat Reactor	BONUS	Boiling water with superheat	Punta Higuera, P. R.	Licensed reactor, construction permit issued	PP-4
High Power Density Reactor Plant	Big Rock Point	Boiling water	Charlevoix, Mich.	Licensed reactor, construction permit issued	50-155
Hallam Nuclear Power Facility	Hallam	Sodium cooled, graphite moderated	Hallam, Neb.	Licensed reactor, construction permit issued	PP-3
Enrico Fermi Atomic Power Plant	Enrico Fermi	Sodium cooled, fast breeder	Lagoona Beach, Mich.	Licensed reactor, construction permit issued but contested	F-16
Experimental Gas-Cooled Reactor	EGCR	Helium cooled, graphite moderated	Oak Ridge, Tenn.	Government-owned reactor, construction permit granted	
Florida West Coast Nuclear Power Plant	FWCNP	CO_2 cooled, D_2O moderated	Pierce, Fla.	Licensed reactor, construction application withdrawn	50-152
Peach Bottom Atomic Power Station	Peach Bottom	Helium cooled, graphite moderated	Peach Bottom, Pa.	Licensed reactor, construction permit pending	50-171
Piqua Organic-Moderated Reactor	Piqua	Organic cooled, organic moderated	Piqua, Ohio	Government-owned reactor, construction permit granted	PP-2
Carolinas-Virginia Tube Reactor	CVTR	D_2O cooled, D_2O moderated	Parr, S. C.	Licensed reactor, construction permit issued	50-144

Table III-1 (continued)

Short Name of Plant	Type of Fuel	Loaded Excess Reactivity	Individual Rod Worth	Total Rod Worth
PWR	Core seed, U^{235}-Zr alloy; blanket, natural UO_2	15%	Not given; 0.625% (av) calculated	20%
Indian Point	Highly enriched UO_2 and ThO_2	12.8% at 450°F	Not stated; can stand loss of two rods at 450°F	Movable, 12.5%; fixed 3% at 450°F
Yankee	UO_2, 3.4% enriched	19.8% in cold, clean condition; 6.9% at equilibrium	1% (max) for center rod; others less	17%; need 25%
N.S. Savannah	UO_2, 4.4% enriched	11.2% in cold, clean condition	1.3% (max), 0.93% (min) in cold, clean condition	15% in cold, clean condition; 18% at operating temperature
Saxton	UO_2, 3.7% enriched	19% in cold, clean condition; 11% in hot clean condition; 7% at equilibrium	~1.5% at equilibrium	Not listed; insufficient for holddown
VBWR	UO_2, 2.3 to 4.5%	12%	Center rod, 5%	18%
Dresden	UO_2, 1.5% enriched	13.1% in cold, clean condition	Not stated; can stand loss of one rod in cold, clean condition or three rods in hot, clean condition	14.1% + 2% for end-connector poison
Elk River	Fully enriched UO_2 + ThO_2 mixture	14.9% in cold, clean condition	4.2% for center rod	16.9% and 4.5% burnable poison in fuel
Pathfinder	UO_2, 1.8% enriched, in boiler; UO_2, 93% enriched, in superheater	13% in cold, clean condition	1.5% for most effective rod	17%
Humboldt Bay	UO_2, 2.6% enriched	Stated to be relatively meaningless because of reactor characteristics (page 30 of ref. 18, April 15, 1959)		
BONUS	UO_2	16.6% in cold, clean condition	1.4% at operating temperature; 1.8% (max) in cold, clean condition	19%
Big Rock Point	UO_2, 3.2% enriched	23% in cold, clean condition	Not stated; can stand loss of first rod out in cold, clean condition	28% in cold, clean condition
Hallam	Uranium-10% Mo alloy, 3% enriched	5.5% at design power plus 0.8% because of fast temperature effects; total, 6.3%	Not given; calculated from totals for safeties, 0.6% each; shims, 0.61% each	Safety rods, 9%; shim and regulating rods, 8%
Enrico Fermi	Uranium-10% Mo alloy, 26% enriched	40 cents	Said to be $1.00 for each of eight safety rods; 46 cents each for two operating control rods	6.5%
EGCR	UO_2, 2.55% enriched	14.88%	Not stated	18.2%
PWCNP	UO_2, 1.2% enriched	0.7%, equilibrium operation	1% for most effective rod	12%
Peach Bottom	Homogeneous solid U^{235} and Th^{232} in graphite	29.3% at equilibrium	Most effective rod, 0.01 ± 20% step if suddenly withdrawn	Not stated; 31.3% at equilibrium
Piqua	Uranium-3.5% Mo-0.1% Al alloy, 1.94% enriched	6.5% in hot, clean condition	7 rods 1.43% each at 300°F; 13 rods 0.94% each	Not given; 10% at 300°F per ref. 5
CVTR	UO_2, 1.8% enriched	36% in cold, clean condition	1.4% (av)	44.8%

Table III-1 (continued)

Short Name of Plant	Shutdown Margin	Auxiliary Holddown Means	Rod Guides	Type of Rod Drive
PWR	5%	None	Guides built into seed fuel elements	Split roller and screw
Indian Point	2.4%, worst case	Soluble poison, always, for cold, clean core	Guides above core and in bottom grid plate; drive rod guided by its seal; spacer buttons on control rods slide along fuel can surfaces	Motor-driven stop and oil-hydraulic cylinders
Yankee	Inadequate by 4.5% in cold, clean condition; adequate in hot, clean condition	Liquid poison required to give 5% cold, clean margin	Full length	Magnetic jack and latch
N.S. Savannah	2.8%	Soluble poison	Guides through and above core	Electromechanical and oil-hydraulic drive with mechanical latches
Saxton	None in cold, clean condition; 2 to 3% at equilibrium	Liquid poison required to hold down cold, clean reactor	Follower rods in upper grid plate, guides below core	Magnetic jack
VBWR	3%	Liquid poison if needed	Fuel channels and guide bushings in vessel head	Electromechanical with pneumatic cylinder and balancing cylinder
Dresden	3%	Soluble poison if needed	Guide tubes below core only, fuel assemblies in core	Water-hydraulic drive with mechanical latches
Elk River	6.5%	Soluble poison if needed	Continuous	Rack and pinion, H_2O lubricated
Pathfinder	4%	None	Tubes through core and above, halfway to drives	Rack and pinion
Humboldt Bay	3%	Liquid poison	Guides below core only, fuel assemblies in core	Water-hydraulic drive with mechanical latches
BONUS	2.4%	Fixed poison rods and soluble poison if needed	Channels above core, fuel-element boxes in core	Rack and pinion, H_2O lubricated
Big Rock Point	5%	Soluble poison if needed	Guide tubes below core only; fuel channels act as guides in core	Water-hydraulic drive with mechanical latches
Hallam	Total, ~10.7% from full power	None	Metal sleeves in graphite moderator	Sprocket and cam for safety rods, magnetic support; ball nut and screw for shim rods
Enrico Fermi	6%	None	Continuous	Lead screw and ball nut
EGCR	3.32%	None	Tube above core, graphite in core	Cable and drum
FWCNP	4%	Dump enough D_2O to get 6% additional	Nonfueled pressure tube	Cable and drum
Peach Bottom	2% at equilibrium	None	Graphite tubes in core rod seal in thimbles	Electromechanical with oil-hydraulic and pneumatic cylinders and mechanical latches
Piqua	3.4% when core is most reactive	None	Full length	Magnetic jack
CVTR	8.8%	None	Core support structure	Rack and pinion, D_2O lubricated

Table III-1 (continued)

Short Name of Plant	Location of Rod Drive	Seals	Scram Direction	Scram Force
PWR	Top of vessel in thimbles	None required	Down	Gravity
Indian Point	Below pressure vessel attached to buffer-seal flange	Buffer seal, sliding shaft	Down	Vessel pressure and gravity
Yankee	Top of pressure vessel in thimbles	None required	Down	Gravity
N.S. Savannah	Top of vessel above thimbles	Buffer, sliding	Down	Accumulated pressure through hydraulic cylinder
Saxton	Below pressure vessel in thimbles	None required	Down	Gravity
VBWR	On movable platform above reactor; all seals and rods go through single plate assembly	Water seal	Down	Air pressure from accumulator to pneumatic cylinder of drive; backed up by motor which drives at high speed to follow up insertion
Dresden	Bottom of pressure vessel in thimbles	None	Up	Accumulator pressure; one for each three rods or reactor pressure to each cylinder
Elk River	Bottom of pressure vessel in thimbles	Rotary	Down	Gravity
Pathfinder	Thimbles above pressure vessel submerged in shield-water pool	Double reactor vessel to seal chain and seal drain to oil-filled drive housing	Down	Gravity and scram springs, if fully withdrawn
Humboldt Bay	Bottom of pressure vessel in thimbles	None	Up	Accumulator pressure; one for each two rods or reactor pressure to each cylinder
BONUS	Thimbles at top of vessel	Rotary	Down	Gravity
Big Rock Point	Bottom of pressure vessel in thimbles	None	Up	Accumulator pressure; one for each rod or reactor pressure
Hallam	Top of biological shield	None required	Down	Gravity, only safety rods can scram
Enrico Fermi	Top	Bellows and metallic "O" rings	Down, safeties only	Gravity, with spring acceleration
EGCR	In top biological shield	None	Down	Gravity
PWCNP	Top of upper shield tank	None	Down	Gravity, with spring acceleration
Peach Bottom	Below thimbles on bottom of pressure vessel	Gas seal, sliding	Up	Accumulator pressure on hydraulic cylinder or gas in pneumatic cylinder; not explained
Piqua	Upper part of reactor vessel	Canned and submerged in coolant	Down	Gravity
CVTR	Reactor header cavity	None	Down	Gravity

Table III-1 (continued)

Short Name of Plant	Opposition to Scram Force Other than Drive and Seal	Scram Time, Objective or Actual	Startup Procedure	Data Source
PWR	Upward water flow	Not given; maximum velocity, 8 fps; ~70 in. rod travel	Four rods withdrawn at one time to start with new core; 12 rods are seated at full power; four rods withdrawn at one time to compensate for aging	Ref. 12
Indian Point	Back pressure of hydraulic cylinder and coolant flow	Two-thirds insertion in 0.864 sec after receipt of scram signal	Below 10^{-6} times full power, one rod or one group of five rods may be withdrawn at a time; above 10^{-6} times full power, rods may be withdrawn, as programed in groups of four or five; above 15% of full power, rods may be withdrawn, as programed, in groups of four or five, manually or automatically	Ref. 20
Yankee	Against water flow	<2 sec after initiation of scram signal	Rods withdrawn in groups starting with outside eight and others in turn; reactor operates with some rods out, some in, and some partly in	Ref. 4
N.S. Savannah	Vessel pressure; 1750 psi	66 2/3% rod insert in 0.8 sec after receipt of scram signal	Manual withdrawal of rods in programed groups up to 15 to 20% power; nine rods may be put on servo	Ref. 21
Saxton	Water flow upward	Not given	One rod withdrawn at a time	Ref. 6
VBWR	Vessel pressure, ~1125 psi	0.3 sec for 33 1/3% full travel	One rod withdrawn at a time	Ref. 22
Dresden	Vessel pressure, 1000 psi, and gravity	10% in 0.60 sec; 90% in 2.5 sec	One rod withdrawn at a time	Ref. 17
Elk River	None	95% in 1.2 sec	One rod withdrawn at a time above 425°F; motion blocked below this temperature	Ref. 7
Pathfinder	Water flow upward	With signal delays, slightly more than 1 sec full stroke	One rod withdrawn at a time by operator "selection to provide optimum utilization of fuel"	Ref. 8
Humboldt Bay	Vessel pressure, 1010 psi		Startup not specified; operate by manual adjustment of control rods; programing to be determined	Ref. 18
BONUS	Upward flow of water in core	Not given	One rod withdrawn at a time	Ref. 10
Big Rock Point	Vessel pressure, 1050 psi, and gravity	2.5 sec for 70% of full stroke	One rod withdrawn at a time	Ref. 19
Hallam	None on safeties	Not given; system delay to effect release in <0.30 sec	Safety rods cocked first; cannot be motor driven to insert when shims are off seats; automatic control operates shims in power range	Ref. 11
Enrico Fermi	Upward sodium flow	~0.6 sec, with 0.3 sec used in getting to core	Rods may be moved singly or as groups; speed reduced once power range is reached; safety rods withdrawn first	Ref. 13
EGCR	None	~3 sec for full travel	All rods withdrawn simultaneously up to low power; then by groups of four or five	Ref. 15
FWCNP	Some pressure drop across rod because of its piston action in the fairly close-fitting pressure tube	<3 sec for full travel	Manual withdrawal of single rods	Ref. 16
Peach Bottom	None	Not given; system delay 0.20 sec	One rod withdrawn at a time; programed at power with servo control; some rods cocked before startup	Ref. 14
Piqua	Coolant	0.07 sec release; 0.800 sec maximum time	One rod withdrawn at a time; manual and automatic switching to make selection	Ref. 5
CVTR	Water displacement in guide tube	Not determined	Not established; possibly four-rod group withdrawal	Ref. 9

Essentially it consists of a number of magnetic circuits placed adjacent to each other along the axis of, and concentric to, a long cylindrical steel rod which attaches to and serves to move the control rod. Part of the magnetic circuit is arranged to move in a direction parallel to the rod under the influence of the fields set up when direct current is passed through the exciting-coil windings. By energizing the several magnet coils in the proper sequence, the rod extension may be moved up or down in short steps or held in any position, as needed. In a heavy-duty mechanism of this type, fingers are added to the movable magnetic members which engage grooves in the lifting rod when the proper coils are energized, and thus effect positive gripping of the rod. Such a drive requires, as auxiliary equipment, a source of direct current and a motor-driven selector-switch assembly for programming the excitation of the magnets. Since all moving parts of the drive are fitted into a thimble that is part of the pressure vessel, no penetrations are required. However, the moving parts are subjected to vessel internal pressures and, to some extent, to its environment. Changes in alignment resulting from temperature and pressure extremes within the reactor do not seem to cause any unusual design difficulties. The rods move vertically upward to add reactivity and downward under the force of gravity to scram the reactor. Springs are used to provide initial acceleration to all rods that have been fully or nearly fully withdrawn. The drives may be mounted either on the top or bottom head of the vessel to lift the rods out of the core from the top or to push them out from the bottom, as the case may be. Coarse position sensors are provided as part of the rod drive, and fine position information comes from a continuous position sensor actuated by the motor-driven selector switch. The rod withdrawal rate is determined by the speed of the a-c motor driving the selector switch; however, the maximum withdrawal rate is limited by the inherent characteristics of the magnetic jack. Some specifications for the Saxton reactor and details of its rod-drive mechanism are being revised. Information on these revisions will unfortunately not be available in time for inclusion in this review.

Rack-and-Pinion Drive Mechanism. The Elk River,[7] Pathfinder,[8] Carolinas-Virginia,[9] and Boiling Nuclear Superheater (BONUS)[10] reactors make use of the rack-and-pinion drive mechanism. In general, the only auxiliary equipment or system required by this arrangement is a means for providing purging and cooling to the pinion-shaft seal. Late information on the Carolinas-Virginia drive indicates that some means have been devised for eliminating the seal, but details are not available at this time. Even if no cooling of the mechanism is required in a particular case, purging of the seal with purified or deactivated reactor coolant is necessary to wash any "hot" coolant or radiolytic gases back into the reactor. In some cases this seal is subjected to full reactor pressure, which may be as high as 900 to 1000 psi. In other cases the pressure is low, but the seal may be required to provide containment for the D_2O moderator, and therefore the seal design must be better than average. Thimbles with extension tubes are used to contain the rack and pinion, and these are subjected to the reactor vessel pressure. A scram is effected by releasing a magnetic clutch through which power is transmitted from the motor to the pinion. When the rods are moved vertically, and they usually are, scramming is under the influence of gravity, with initial acceleration provided by springs in some cases. The mechanism inside the thimble or other vessel extension is lubricated by water, usually at some elevated temperature. Water is not a particularly desirable or satisfactory lubricant, but its use is convenient and, in fact, is not easily avoided in water-cooled reactors. Position information is usually taken from the pinion shaft through suitable gearing, which gives the position of the rods at all times as long as the mechanical system remains intact and in adjustment. Insert and withdraw limits are detected by switches actuated from the gear train used for presenting continuous position data.

There is a situation concerning both the magnetic-jack and the rack-and-pinion types of drives which, although no doubt recognized by control-rod-drive designers, should be mentioned for the benefit of others. Since the extension rod from the drive to the absorber passes through the vessel wall, it could become a piston if the thimble enclosing the drive failed while the reactor was pressurized. The thimbles and holddown means are obviously designed to operate under such pressures, and no trouble is anticipated under design conditions. Suppose, however, that the covers or thimbles were not

properly replaced and the reactor was brought up to temperature. The fluid contents of the vessel escaping past the extension rod and out the cover or thimble flange would tend to blow the rod out and produce an unbalanced force on the drive which it probably was not designed to withstand. The concern here is the probable results of a scram being called for while the pressure is high, for the case of the rod drives mounted above the reactor. Instead of the rod falling into the core when the magnetic clutch is released, it would be shot out of the core by the piston action.

Nut and Lead-Screw Drive Mechanism. The nut and lead-screw drives reviewed (Hallam, PWR, Enrico Fermi, and Peach Bottom) have little in common. In the Hallam reactor,[11] this type of drive is used only on the shim rods. These are not scrammed and are operated in thimbles that extend down from the top of the reactor through the core. Contact with any sodium or vapor from the reactor is prevented by sealing the thimbles and then pressurizing them with helium. This necessitates the use of an auxiliary helium system. The rod is attached to its drive through a disconnect mechanism that is accessible from the loading face. All drive mechanisms are apparently supported by a common carriage that must be moved to permit access to the reactor. It is not clear whether there is a positive arrangement that prevents the carriage from being moved until all rods are disconnected. Continuous position sensors are provided, but travel limit sensors are apparently omitted.

The scram function in the Hallam reactor is vested in a separate set of rods. These are supported directly by magnets which, upon releasing, allow the rods to fall under the influence of gravity. The magnets are, in turn, supported on roller chains, with the result that no force can be applied to insert any rod that might stick. These rods are also housed in sealed, helium-pressurized thimbles, and their mechanisms likewise operate at approximately room temperature. They are mounted on the carriage with the shim rods, and the whole assembly is moved as a unit, when necessary, as noted above. Sensors are provided which are said to be able to detect when the safety rods are seated or are in contact with their lifting magnets and to detect the position of the magnets. If these sensors are independent and receive their information directly from the proper sources, they constitute a useful backup means for each other. The pressurized thimble is effective for this application, but it probably is limited to use in low-pressure reactors.

The Shippingport PWR[12] is a high-pressure type of reactor that avoids shaft seals by using canned-rotor motors. Each control rod is individually driven through a long extension rod by a type of nut-and-screw device. The nut is a two-piece roller arrangement driven directly by the rotor of the motor. The magnetic field set up by the motor windings is used to keep the nut halves in contact with the screw for holding or moving the control rod. To scram a rod, the motor field is deenergized to release the nut halves, and the rod falls by gravity. The motor rotor, as well as the other moving parts of the drives, operates in water at reactor vessel pressure.

Two auxiliary systems are needed with this drive. One of these provides cooling for the motors and drive mechanisms and consists of suitably arranged cooling coils and associated water flow supply and control means. The interior of each pressure thimble contains a heat barrier assembly, and the thimble is made long to increase the thermal path from the drive to the pressure vessel. The other auxiliary system is needed in connection with the motors. For both mechanical and electrical reasons, the canned-rotor motors are operated on low-frequency alternating current and, when the motors are holding the rod, they must be excited with direct current. The electric power inverters, power sources, and control systems for the rod drives constitute a sizable installation. Rod speed is ultimately determined by the speed of d-c motors, since they are part of the inverter system. Although the upper speed limit of a d-c motor is not well defined, it is stated that the d-c motors used in the PWR installation are incapable of driving the inverter brushes appreciably above normal speeds because of their limited horsepower. This limitation would not exist, however, if, as an operating expediency, a defective motor were replaced with one having higher horsepower. Position information is apparently provided by a system of electrical windings arranged in the same general manner as those used with the magnetic-jack drive. The position of the rod is known as long as the device functions, provided, of course, that no mechanical failure occurs, such as a broken screw or

drive-extension rod. Rods may be uncoupled from their drives by remote means, when necessary, so that they remain seated in the core during operations such as refueling. Control-rod holddown during refueling depends upon an arrangement that must be fastened into the reactor vessel after its head has been removed. The referenced report did not mention whether any means have been provided for preventing the rods from being lifted along with the vessel head.

A special feature of the Enrico Fermi reactor[13] is that it does not require a split nut to permit scramming. Each rod is coupled to its drive by means of a magnetically held latch that is released (by turning off the magnet current) to drop the rod. This arrangement permits the rods to be dropped at any time independent of, and regardless of, the action or failure of the normal drive. Many research reactors have successfully employed electromagnets to support the control rods either directly or through latches, as is done in the Fermi drives. This latter method permits the electromagnet to be located in a relatively low-temperature zone of the reactor. As with almost every mechanism, there are some design problems which are peculiar to that mechanism. One difficulty is to provide a permanent seal between the electromagnet, its armature and rather long latch-release rod, and the sodium atmosphere within the reactor. Sealing is through a bellows, which is not in an accessible position for inspection and testing once the reactor is put into operation. The designers feel sure that, even if the bellows leaks and sodium or its oxide vapors should find their way up into the close-fitting guides of the armature and latch-release rod, there will be no freezing of the vapors once the reactor is put into operation because the temperatures will be too high. However, the original designers will have lost control of the situation when the reactor operators take over their normal responsibilities and, in the natural course of events, the reasons for maintaining proper temperature will be forgotten. There would follow a freezeup that would be totally unsuspected by the operators. The control worth is large compared with the loaded excess reactivity (about 8 to 1 or more) so that the failure of several rods to scram will not prevent shutdown and holddown.

The six safety rods are arranged to move vertically, not only in a slow-speed mode for normal reactor operation but also in an overriding high-speed, insert only, mode. Two motor and speed reducer trains are used with each rod drive. The high-speed insert drive is designed so that it is incapable of withdrawing the control rod regardless of the direction of rotation of its motor. Presumably, fast insertion is automatically called for in scram conditions, and therefore use may be made of the high-speed insert drive as a backup for latch failure or sticking rods. These safety rods are held in their fully withdrawn positions during normal operation. Energy for their initial acceleration on scramming is provided by individual helical springs and gravity. The energy is restored to the springs by the drives during rundown following a scram. Gravity would, however, keep the rods seated if their drives failed to follow.

The two operating control rods are attached directly to their drives and are therefore not used for scramming. Position information is taken from the drives of both types of rods by means of multiturn potentiometers and position switches, but the actual position of the safety rods is known only so long as they are attached to their drives. No means are provided for determining directly whether the safety rods are seated or coupled to their drives, but this is not surprising, considering the operating environment. The drive mechanisms are sealed from building atmosphere and operated in a dry inert gas. The rod extension seal is of the metallic O-ring type. An inert-gas system is required in order to maintain the desired atmosphere within the drive assembly housing.

A quite different arrangement is used in the ball-nut and lead-screw drive for the Peach Bottom reactor.[14] In this case, power must be used, and the drives must function to effect any and all shutdowns. The absorber rods are inserted from below, against gravity, and the drives remain coupled to the control rods at all times, since part of the mechanism constitutes the bottom in the core. No scram in the usual sense of the word is provided since the fast insert which is provided is capable of exercising suitable safety action. The normal and high rod speeds are obtained with a single hydraulic motor that operates at two speeds, and not by using two separate electric motors and gear trains, as was done with the Fermi drives. It is necessary, of course, to avoid inadvertent or accidental high-speed withdrawal, and the method here is to use different energy sources

and separate valves for control. In normal operation, hydraulic fluid from a common controlled-pressure source is fed through individual isolating check valves, flow-limiting fixed orifices, and direction-control valves to the piston type motors. The valves are electrically actuated and spring closed, and therefore the motor is shut off on loss of its control power.

The fast-insertion system consists of an accumulator type of pressurized fluid supply and a second flow control valve. One of these is built into each power unit to secure independence of action. The control valves are spring opened and solenoid closed so that loss of control power is intended to result also in fast insertion of the control rods. Bottled nitrogen is used to pressurize the hydraulic fluid in the accumulator through a floating piston. A substantial amount of instrumentation is employed to help ensure that sufficient oil is always available at the required pressure. Unsatisfactory conditions and instrument failures are said to initiate insertion of the affected rod. Although a sudden loss of pressure would, of course, prevent rapid insertion, it would not be likely to interfere with normal-speed operation unless there was a major rupture in the fluid system. It would be reasonable to assume that such failures are only remote possibilities and, furthermore, that they would probably not occur simultaneously in several or all drives.

Hydraulic fluid for filling the accumulators also comes from the system used to power the drives for normal operation. Individual drives may be isolated by closing associated electrically operated valves or manual backup valves, when necessary. Under all normal conditions, dependence is placed on check valves to provide isolation, and thus prevent possible interaction between drives. It seems particularly important that the check valve protecting the accumulator from back-feeding to the supply system operate properly at all times. Suppose a rupture occurred in the hydraulic fluid supply system and thus precipitated a shutdown demand. If any of the check valves failed to function properly, it is conceivable that the affected accumulator might run out of oil before the associated rod had been inserted. It is assumed that the accumulators and their filling lines are sized so that, even if a line should rupture or the check valve should seize in the fully opened position, a complete insertion of the associated control rod will occur on demand.

The control rod is not a rigid structure, and in one proposed design it is made up of short sections coupled with knuckle joints. The absorber material is in the form of boron-loaded graphite cylinders that are fitted over central support rods; these are joined by the knuckles. This articulating arrangement is necessary to circumvent the effects of differential thermal expansion. In this design, contact between the uncanned absorber and the graphite guide tube is prevented by making the outside diameter of the absorber somewhat smaller than that of the knuckle joint. Mechanical considerations make it necessary to limit the downward acceleration and deceleration to less than one G. To this end a spring-opened throttle valve placed in the exit oil line of the motor is coupled to the drive in such a manner that it reduces the oil flow progressively over a short portion of the rod travel as the rod moves toward its full inserted position, thus providing the required deceleration to terminate fast insertions.

The nut and lead-screw drive is of the low-friction type, and, although there is no unbalanced force on the rod because of vessel pressure, there is a decided tendency for it to move downward under the force of gravity, especially if there is oil leakage from the motor or if it is uncoupled to permit removal of the drive. This is a potentially dangerous situation because downward motion of the rods is in the direction to increase reactivity. Dependence is placed, in this case, on a special type of clutch designed to transmit rotary motion from the motor to the rod but not in the other direction. This clutch then is the bottom for the reactor core, but the reviewer found no mention in the reference documents of any testing procedure or schedule to be followed to ensure its proper functioning. Presumably the operator would notice the drift on the associated rod position indicator during operation, but it is not so clear how movement would be discovered when the reactor was in the shutdown condition, for example, since it would not be considered necessary to watch the position or other indicators.

The nut and lead-screw assembly and its long push rod are housed in a pressure tube attached to the bottom of the reactor pressure vessel. The tube is purged from near its bottom, and the helium flows up and around the nut and lead screw and push rod. The hydraulic drive unit is mounted below the lead screw in a removable section of the pressure tube. A shaft seal is

provided on the lead-screw shaft just above the clutch. This permits removal and replacement of the hydraulic power unit assembly or inspection of the clutch without depressurizing the reactor vessel. In operation, this lower section is also gas-pressurized but is not purged.

It is not quite clear what force the hydraulic motor is capable of exerting against a stuck rod. Since the guides are graphite, the force is presumed to be low, and on this basis the drive is probably incapable of damaging any of its mechanical parts should be control valve seize in the open position. It is stated, however, that the motor is incapable of damaging the gear train should this become locked.

Because the control rod and its drive are rigidly coupled, it is a simple matter to provide rod position information by gearing suitable sensors to the hydraulic motor output shaft. Continuous-position and travel-limit sensors are said to be included in the present design.

Cable-and-Drum Drive Mechanism. The cable-and-drum type of rod-drive system is used in two reactors covered in this review, the EGCR[15] and the Florida West Coast Nuclear Power Plant (FWCNP).[16] In the EGCR the unit is fabricated in such a manner that the drive, its housing, the rod guide tube, radiation shielding, and the rod constitute a single plug type assembly that can be installed or removed intact. All rods are withdrawn simultaneously to effect a startup. In the region of low-power operation, temperature rate-of-change limitations necessitate withdrawal of rods by groups of four or five, but the over-all pattern of withdrawal is maintained. This leaves the rods situated at all times so that they may exercise their maximum capabilities in scram situations. The nuclear heat developed in the rods is removed by flowing cool helium through the guide tube and over the rod. The thimbles, which contain and support the drive-mechanism assembly, are attached to the top of the reactor pressure vessel and open into it to admit the rod and its guide tube. Although the vessel pressure is not expected to adversely affect the operation of the mechanism, heat sinks, heat barriers, and a flowing coolant must be provided to protect it from the reactor temperature. Double use is made of the rod-cooling helium by routing it through the drive mechanism on its way to the rod. The primary coolant is an ineffective radiation shield, so means are provided to prevent streaming through the rod guide tubes and thimbles. Shielding in the form of high-density concrete is built into the rod-drive plug assembly.

The rod lifting cable passes through the concrete shield section at a suitable angle guided by idler pulleys. The lifting-cable takeup drum is driven by an instantly reversible single-phase capacitor start-and-run motor through reduction gearing, a magnetic clutch, and an eddy-current speed limiter. To scram the rod, the clutch is deenergized, and gravity drives the rod into the core. The system is relatively slow since several sheaves, the cable drum, a jack screw, the eddy-current speed limiter, several gears and pinions, and the clutch all have to be accelerated by the rod weight, which, in addition, must overcome the friction losses in the system. Continuous-position, travel-limit, and two intermediate-position sensors are provided. They all take their information from the drum shaft and not from the rod. A slack-cable sensor is also included, but this does not provide any guarantee that the operator can actually tell that the cable has failed and the rod has dropped. Although there is a positive bottom in the core that would prevent a rod from falling through if a cable broke, there is no way to apply any force to insert a rod that is stuck. At least one alternate drive mechanism is being considered for this reactor. It is of the same cable-and-drum type, but it differs in the method of providing shock-absorber action, as well as in physical arrangement. Since there is no indication at this time whether this latter design will be selected, it will not be considered further in this review.

In the most recent design of rod drive for the FWCNP reactor, the mechanism is arranged as a package assembly that mounts on an otherwise unused fuel pressure tube. The tube serves as a guide for the rod, and it can also be considered the means for providing a bottom in the core should the cable and the emergency buffer both fail. The rod is quite long, extending completely through the upper shield tank in all positions, and the rod and the buffer assembly provide all the shielding necessary to prevent flux streaming from the tube. Heat removal is not a special problem, as in the case of the EGCR, and a continuous purging system is not needed. Further, since the mechanism housing is not open to the reactor atmosphere, the probability of the drive becoming contaminated is very small.

Control rods are to be withdrawn individually in startup so that, at all times, some rods are fully withdrawn. Because of the position of these rods and the mass of the rod, drum, cable, clutch, brake, sheaves, and other parts of the drive, the response in scram situations leaves something to be desired if gravity alone provides the insertion acceleration. As is the case with many other designs, this one makes use of a helical spring to supply the required additional initial acceleration. Extensive use has been made of antifriction bearings, splines, etc., to minimize friction in the mechanism.

Further control of the action is provided by adjusting the ratio of full fall travel and braking travel. Although it is intended that the rod seat on the buffer, it is the experience of the reviewer that suitable adjustments are difficult to make and more difficult to hold. Some slight undertravel is usually accepted to avoid a slack-cable condition. From the control standpoint, the seated position of the rod probably does not have to be fixed so definitely that mechanical stops are needed, except as backups to provide the bottom in the core.

Five sensors are included within the drive mechanism assembly. Four of the sensors have counterparts in the EGCR and furnish information of the same class. These are the two travel-limit switches, the slack-cable switch, and the continuous-position synchro. The fifth sensor is a position switch that is actuated directly by the control rod when it reaches its fully withdrawn position, rather than through gearing from the cable-drum shaft. The information from this sensor is more accurate than that otherwise available and may be used in checking the information from the other sensors. Since the cable will probably stretch in time, compensating adjustments of other sensors will be necessary.

Fluid Drive Mechanism. Hydraulic systems that are similar are used in the Dresden,[17] Humboldt Bay,[18] and Consumers Power (Big Rock Point)[19] reactors. The drives are mounted inside the lower ends of long thimbles on the bottom head of the pressure vessel, and no shaft seals are required. They are pressurized from the reactor feed-water system for normal use and from accumulators or reactor pressure when scramming. The absorber rods are arranged to scram upward into the bottom of the core. As described, the hydraulic cylinder pressurizing system is maintained at some minimum pressure above that of the reactor vessel under normal operating conditions. If this differential falls sufficiently or is lost entirely, full reactor vessel pressure is said to be automatically transferred to each hydraulic cylinder mechanism by means of a self-contained, self-acting ball or similar type of valve. Because of the differences in opposing pressure areas of the control rod and the drive, the net force available upon loss of normal working pressure is great enough to scram the rod, at least as long as the vessel pressure exceeds a few hundred pounds per square inch. In any case, scramming any rod depends upon the proper operation of two or more devices in sequence, and the potential for inducing failures is therefore present.

Since there may be some misunderstanding at this point, a few words of clarification are in order. As an example, consider a rod drive in which the rod is coupled directly by an electromagnet (no mechanical or other latches). Scram is effected simply by cutting off the magnet's current, so that any number of devices may be employed to act in such a way that each directly and independently cuts off the current and therefore produces a scram. Consider next a rod drive that must act to effect a scram, such as a hydraulic cylinder drive that is permanently coupled to the rod. For the cylinder to operate, there must be a control device, which has to be told of the hazardous condition and which must then act to admit hydraulic fluid to the proper side of the piston and exhaust it from the other. Therefore the control device, assuming it is an electrically closed valve or pair of valves, must be cut off by some means, and the valves must open and the hydraulic cylinder must function, all in sequence, to insert a particular control rod. This also presupposes that suitable pressure is available to operate the drive. Thus it is clear that in the first type of rod-drive system, any one of a number of devices is able to directly and independently scram a rod, whereas in the second case several devices have to work in sequence every time to get the scram. In at least one rod control drive reviewed, the sequence included a pneumatic system between the electrical scram signal and the hydraulic valve.

Such arrangements not only make it more difficult to get the desired scram on demand but they also increase the possibility and probability of getting spurious rod insertions because of the number of devices and power sources involved. It is not intended by these observations that hydraulic rods should be discredited as a class, but it is intended that recognition should be given to the number of possibilities for the occurrence of failures and misoperation that do exist in some of the presently available designs.

Under normal operating conditions, the hydraulic drive is intended to place and hold the rod wherever required. However, since hydraulic cylinders leak slightly, the rods tend not to hold their position but to drift slowly in or out, depending on the direction of the residual force. In the systems being reviewed, the drift was apparently arranged to be in the outward direction under operating conditions; therefore a single mechanical latching mechanism could be used to limit the amount of drift, as well as to oppose any pressure unbalance that might tend to blow out the rod if the normal balancing force should be lost. The latch complicates the design, since means must be included for automatically defeating it when a legitimate withdrawal request is received. A question remains concerning the action of a rod, when it is instructed to withdraw, for which a pressure unbalance exists and which is being held in check by the latch. Will the latch-defeating mechanism, necessary for the withdrawal, permit the rod to be blown out? It must be emphasized that this could initiate an incident rather than merely a failure to scram.

The hydraulic pressurizing and control system for these drives is somewhat involved. It makes use of electric, pneumatic, and hydraulic devices, the failure of any one of which will affect or prevent normal operation of the associated drive. There are 80 separate systems in the Dresden reactor, one for each rod, and thus it can be anticipated that one or more systems will be inoperative frequently. "System" in this case is intended to include the electric/pneumatic and pneumatic/hydraulic regulating and check valves, as well as associated apparatus provided directly and specifically for controlling the application of hydraulic fluid to actuate any one rod drive. Failure of, or perturbations in, the electric, air, or water systems could apparently have an effect on reactor operation through interference with several drives simultaneously. The over-all system complication is conducive to the appearance of "made" failures. For example, one or more of the hand-set rod-speed-control valves could be inadvertently opened and thus provide the means for producing unexpected power excursions. The unusual concern about the possibility is pointed up by considering it in relation to rod programming. Where withdrawal is limited to one rod at a time, it is quite likely that the rod speed has been selected so that, on the average, any one rod will add reactivity at about the maximum rate permitted by the reactor design. Increasing the withdrawal speed of any rod would, therefore, permit reactivity additions at excessive rates. Position sensors are not described in the documents, but they are said to be capable of indicating rod position at 8.9-in. intervals in their travel in the case of the Dresden reactor or 3-in. intervals in the Humboldt Bay and Consumers Power reactors.

The Consolidated Edison[20] drive mechanism is a combination of hydraulic and electromechanical elements. The control rods are attached rigidly to, and are moved by means of, push rods that extend outside the reactor vessel through sliding seals. Depending on the pressure within the vessel, there may be some considerable force on the push rod tending to eject it from the vessel. The drive, especially under normal operating conditions, will be required to work against this force to control the position of the rods, and, furthermore, it must be capable of scramming the rod when necessary whether the reactor pressure is up or not. Although the control rods are driven from below, they are pushed upward out of the core for withdrawal rather than downward so that any vessel pressure aids gravity in scramming or keeping the rods seated.

The rod drive consists of a hydraulic oil cylinder, which is attached to the push rod and supplies a force in opposition to that due to vessel pressure, and a motor-driven stop, which limits rod motion only in the withdraw direction. Mechanical contact between the stop and the push rod is through a heavy collarlike member fixed to the push rod. Since the stop is between the reactor vessel and the collar on the push rod, the two are kept in contact as long as the unbalanced force on the rod is toward the reactor. Rod withdrawal is effected by moving

the stop upward toward the reactor. The cylinder pressure forces the push rod along to maintain contact with the stop. In order to insert the rod under controlled conditions, the stop is driven downward to force the push rod to follow by acting on the collar. In order to scram the rod, the pressure in the cylinder is relieved, and the vessel pressure then drives the rod to its seat. In scrams, the motor-driven stop and the push-rod collar become separated, but a switch on the stop detects this and runs the stop in to reestablish contact. The stop prevents subsequent and possibly rapid withdrawal of the rod should cylinder pressure return to normal for any reason, provided, of course, that contact has been reestablished. Note that failure in the hydraulic system is most likely to result in a scram, and not in rod blowout. This arrangement also provides a definite bottom in the core as far as the control rods are concerned. No external force need be provided or latches set to counteract any internal force tending to unseat the rod, as in the case of the Dresden drive, for example. However, it is conceivable that one or more rods might be inadvertently withdrawn during maintenance on the drives while the pressure within the reactor vessel was down. Presumably the hydraulic cylinders will be depressurized before any work is permitted on the stop mechanism. If, however, a stop is moved from the fully inserted position of the drive and the hydraulic cylinder is not depressurized or is unintentionally repressurized, the rods will be withdrawn as far as the stop will permit.

One of the auxiliary control systems necessary with any drive of this type serves to regulate the pressure differential that exists between the hydraulic system and the reactor vessel. Other common control systems include the buffer-seal water system, the hydraulic oil system and its accumulators, the drive-motor power system, and the electropneumatic or electrohydraulic control system that operates the drive. Motors of the two-phase type are said to be used in this reactor to provide variable speed for the drives, and this entails a further complication in the form of a "power controller." The position sensors include one for continuous position information on the motor drive, two travel limits on the rod drives, and one for contact between the rod and motor drive.

The control-rod-drive systems considered thus far have been designed for use with reactors whose location is considered fixed, although there is some doubt that full advantage was taken of that fact in many instances. The situation is quite different in the case of the N.S. *Savannah* reactor.[21] Here the control must function as the ship rolls, for example, and must be able to scram and hold down the reactor even if it is turned over. Also, the N.S. *Savannah* must be able to maneuver, which means that the rod drives and their auxiliary and control systems have to cope much more frequently with wide and rapid load fluctuations than would be the case for the same system installed in a stationary power station. The original rod-drive system provided for this reactor is generally similar to that used in the Consolidated Edison installation, as far as components and auxiliary systems are concerned. However, the rods are operated through the top head of the pressure vessel rather than through the bottom. Vessel pressure tends to withdraw or push the rods upward, whereas the hydraulic oil cylinder, the motor-driven stop, and an additional element (a mechanical latch) each oppose this. Scrams are effected against vessel pressure by the hydraulic cylinder. The latch is intended to hold the control rods in their seats should the cylinder pressure fail before the motor-driven stop follows up after a scram or if the stop fails to follow at all. On the basis that gravitational forces are difficult to make use of for protecting a shipboard reactor, it is probably necessary to live with some system complications. On the other hand, cognizance must be taken of the fact that, although continuous operation of the reactor is of critical importance, the reactor is in a ship, and the shipboard maintenance facilities are limited. The N.S. *Savannah* control system is more complicated than its land-based counterparts, and it is hoped that it will not require an excessive stock of spare parts, specially trained maintenance personnel, or excess maintenance time to keep it functioning satisfactorily. Position sensors provide information on travel limits and the position of the motor-driven stop. The means selected for the latter is a single-turn synchro.

The final drive system to be considered is that of the Vallecitos reactor.[22] It is a variation of the combination drives already described and apparently has similar capabilities and limitations. A double-acting pneumatic cylinder

is used (instead of a hydraulic cylinder), and operating air pressure on the bottom of the air cylinder pushes the rods up and against the motor-driven stop for positioning. In addition to the double-acting pneumatic cylinder, the drive system has another pressure cylinder that is tied to the reactor vessel directly (and pressurized thereby) for the purpose of counterbalancing the vessel forces on the rods. As with the other drive systems, pressure changes in the drive cylinder cause the rods to scram downward, but, in this case, the speed of the d-c drive motor for the stop is increased so that followup takes much less time than in the case of a-c systems. The use of two-speed d-c motors introduces an additional potential for generation of an incident, since it is possible for the electrical power or control system to be inadvertently rearranged during maintenance operations to interchange these speeds. Travel-limit sensors are provided for the rod drives, and a digital type of unit is fitted to the motor-driven stop to give its position to $\pm 3/64$ in.

(A. E. G. Bates)

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Section IV

PLANT SAFETY FEATURES

Removal of Contaminants from Gas Streams

Two comprehensive reviews in *Nuclear Safety*[1,2] have described the removal of fission-product activity from gas streams leaving reactors and fuel-processing plants. These articles included information on the removal of iodine, ruthenium, particulate matter, and noble gases from gas streams, as well as on the design of multipurpose fission-product removal systems. This review presents recent developments in iodine and ruthenium removal, the absorption of nonradioactive contaminants from off-gas streams, and the design of several new cleanup systems.

Recent Developments in Iodine and Ruthenium Removal

Radioactive isotopes of iodine and ruthenium which contaminate the gaseous wastes from chemical processing plants are difficult to remove because of the various states in which they may appear in off-gas systems. Iodine can be present as iodine vapor; as a particulate solid, such as NaI; or as a solute dissolved in entrained liquid droplets or adsorbed on the surface of aerosol particles. Iodine also tends to deposit on the surfaces of off-gas ducts at rates which depend on the concentration and which may be temperature dependent.[3] Ruthenium can be present as the volatile RuO_4, or particulate ruthenium oxides, or dissolved in entrained liquid aerosols. Browning[1] surveyed the use of caustic scrubbers, heated silver nitrate beds, silver-plated wire beds, activated charcoal, and dry soda lime for iodine removal and the use of various adsorbents for volatile ruthenium.

Taylor[4] has studied the rates of iodine vapor absorption in water and in aqueous solutions of sodium hydroxide, sodium thiosulfate, sodium tetrathionate, and sodium sulfate. With a disk type laboratory absorption column, the rates were found to be completely gas-phase controlled for the sodium hydroxide and sodium thiosulfate solutions; appreciable liquid-phase resistance to transfer was found with the water, sodium tetrathionate, and sodium sulfate absorbents. The gas-phase absorption rates were quite rapid, and a correlation with ammonia-water absorption data was obtained.

Several recent investigations emphasize the role of iodine adsorption on aerosols in reducing the efficiency of caustic scrubbers. May and Morris[5] have conducted a series of iodine-absorption efficiency tests on the British BEPO reactor caustic scrubbers. The system comprises four 3-ft-diameter towers, in parallel, containing 6-ft beds of 1-in. Raschig rings. A 5 per cent sodium hydroxide solution was used to scrub 3000 cfm of air containing injected I^{131}. The predicted decontamination factor was 100, assuming perfect liquid distribution, and was 10 to 20 for the expected liquid distribution. The decontamination factors actually obtained for normal operation varied between 29 and 32. Substitution of sodium carbonate solution for the caustic did not change the efficiency. However, when aerosols were introduced into the inlet air stream, the decontamination factor dropped markedly. In one test, in which the air and iodine vapors were passed over a hot tungsten wire heated to 900 to 1000°C (a metal fume generator) before injection into the air feed, the decontamination factor dropped to 10.5. In another test the introduction of lead oxide aerosol reduced the decontamination factor to 1.4. The decrease in efficiency was at-

tributed to the adsorption of iodine on aerosols that were not removed by the scrubber.

Chamberlain and Wiffen[6] demonstrated that iodine in air at a concentration of 10 $\mu g/m^3$ was rapidly and irreversibly adsorbed on lead fume but that the degree of adsorption on 0.1-μ condensation nuclei was small for the same iodine concentration. Experiments performed with the reactor containment shells and air-cleaning systems of the British DIDO and PLUTO reactors[7] demonstrated the relative iodine-removal efficiencies of a sodium carbonate scrubber, a copper knitmesh bed, and a charcoal bed in series with a Vokes filter. Only the activated charcoal bed–Vokes filter combination gave a reasonably high decontamination factor after the initial 1-$\mu g/m^3$ iodine concentration had dropped below 0.03 $\mu g/m^3$ as a result of adsorption on the shell and reactor walls. Since the Vokes filter (efficient down to 0.1-μ particles) did not improve the efficiency of the scrubber or copper bed, the adsorption of iodine on nuclei from the air was not considered to be the cause for their poor performance. The high efficiency of the charcoal bed led to the hypothesis that a gaseous compound of iodine was formed, perhaps from reaction of iodine with trace impurities in the air, which was strongly adsorbed on the charcoal but not removed by the scrubber or copper bed.

Cederberg and Bower[8] found that particulate activity generated by the RaLa off-gas scrubber at the Idaho Chemical Processing Plant (ICPP) decreased the effectiveness of the charcoal beds used downstream of the scrubber for additional cleanup from radioactive iodine. Apparently droplets of entrained sodium hydroxide–sodium thiosulfate scrub solution plus sodium carbonate and sodium iodide were dried and carried through the charcoal bed, ending up on the filter. The decontamination factor for the beds was quite low—between 10 and 30. Substitution of a 5 per cent HNO_3–0.001*M* $Hg(NO_3)_2$–0.001*M* $HgNO_3$ scrub solution for the caustic in the scrubber apparently decreased this entrainment problem. This was demonstrated by a large decrease in the activity fed to the charcoal beds. The mercuric and mercurous ions were added to prevent volatilization of iodine from the scrub solution.

The design of silica-gel adsorbers for removal of volatile ruthenium from ICPP waste calciner off-gas has been described by Edwards.[9] The system includes four 7-ft-diameter silica-gel (6 to 12 mesh) adsorption beds in depths varying between 36 and 62 in. Three beds are on stream at one time, while the fourth is being regenerated by a water-washing technique. The problems of dust accumulation in the beds during operation and of decrepitation of the silica gel during regeneration have been investigated.

Removal of radioactive volatile ruthenium from waste calcination off-gases has also been studied at Harwell.[10] Adsorbents tested included silica gel, ferric oxide, kaolin, fuller's earth, and molecular sieves. A proposed calcination pilot plant will use a small disposable silica-gel or ferric oxide bed for cleaning the off-gas from each batch of calcined waste. A decontamination factor of about 10^4 is predicted for volatile and nonvolatile ruthenium across the combined adsorber and filters. A Hanford study[11] demonstrated greater than 99.9 per cent removal of both types of ruthenium, with a combination silica-gel bed and glass-fiber filter. The particle sizes for the ruthenium aerosol varied between 0.5 and 0.1 μ. This study also demonstrated that the adsorption of volatile ruthenium on activated charcoal can be very hazardous. About 200 mg of RuO_4 adsorbed on a 1-g charcoal bed caused the bed to "explode with a flash of flame," possibly as a result of the decomposition of RuO_4 to RuO_2, which released oxygen for the combustion with activated charcoal.

Removal of Nonradioactive Contaminants

In radiochemical processing, it is frequently necessary to remove the nonradioactive contaminants in the off-gas wastes upstream of the radioactive cleanup equipment such as adsorbers and filters. For example, acid gases, such as CO_2, HCl, HF, F_2, SO_2, and Cl_2, and the oxides of nitrogen must be absorbed and neutralized to prevent corrosion of the filters and blowers. Adsorber capacities may also be improved, as shown by Browning et al.,[12] who demonstrated the effect of small amounts of carbon dioxide on the capacity of charcoal beds for adsorbing krypton gas. In addition, prior removal of a fraction of the liquid and solid particles will decrease the loading and improve the life of the filters.

Caustic scrubbers are frequently used for acid gas absorption,[13] and they may also pro-

vide a degree of particulate separation. Packed towers of reasonable packing depth can remove more than 99 per cent of the CO_2, HCl, Cl_2, or SO_2 using 5 per cent NaOH as the scrub liquid. However, the efficiencies for the removal of the oxides of nitrogen with dilute caustic solutions or water are much lower.[14,15] Complete absorption of HF and 98.4 per cent absorption of F_2 by 5 per cent KOH solution in a spray tower has been demonstrated by Liimatainen and Levenson.[16]

The efficiency of entrainment separation by scrubbers is dependent on the inlet gas particle-size range and the design of the scrubber. Silverman[17] estimated that the average acid mist particle generated by sparging or dissolving would have a 20-μ diameter. Nichols[18] derived a relation for predicting the particulate concentration of an on-off gas stream and estimated a 10-μ mean particle size for superficial off-gas velocities of less than 0.15 ft/sec. However, after several changes in direction in an off-gas line, the mean particle size was estimated to be about 3 μ.[19] Other data obtained by Schlea and Walsh[20] and by Garner et al.[21] during evaporation experiments substantiate these estimates. The deentrainment efficiencies of many types of wet scrubber have been determined.[17,22] In general, a wetted, packed, fiber-bed scrubber is more efficient than a packed tower and will remove particles of much smaller diameter. Manowitz et al.[23] determined that a 4.5-ft glass-fiber bed was more efficient than a 13-plate bubble-cap column and that an 8.8-ft Raschig-ring column was less efficient than the bubble caps for deentrainment service in an evaporator. Schlea and Walsh[20] developed an excellent impingement cap that could be used for deentrainment in the vapor outlet of a scrubber.

Design of Cleanup Systems

Designs of several off-gas systems at ORNL have been described in *Nuclear Safety*[1,2] and elsewhere.[24] Ruch[25] recently described a six-stage spray-nozzle scrubber for removing fluorine and hydrogen fluoride from 3000-cfm ventilation air passing through the Volatility Pilot Plant at ORNL. The scrub liquid is 5 to 10 per cent KOH circulated at a maximum rate of 180 gal/min. The scrubber is 4 by 4 by approximately 25 ft long and contains a deentrainment section of baffles and a Monel-fiber demister.

The ORNL F3P plant[26] is currently using two 24-in.-diameter 12-ft-high scrubbers packed with 6 ft of graded 1-lb and 1.5-in. Intalox saddles. The scrubbers are designed to clean 450-scfm air containing ammonia and acid gases. The first scrubber uses 4*N* HNO_3 to remove the ammonia, and the second scrubber scrubs the acid gases with 8*N* NaOH. An off-gas heater and two parallel filter chambers are installed downstream of the scrubbers. A design study of an off-gas caustic scrubber for the ORNL Power Reactor Fuel Reprocessing (PRFP) Pilot Plant[27] indicated that 1.5-in. stainless-steel "Pall Ring" packing would have a very low pressure drop and a high acid-gas absorption efficiency and would be free draining, which would simplify decontamination of the equipment. (J. M. Holmes)

Analytical Radiochemistry and Nuclear Incidents

Analytical radiochemistry is indispensable in the investigation of nuclear incidents, as well as in environmental monitoring of atomic installations. This review cites some past incidents to show the role of radiochemistry in emergency situations involving the release of radioactivity. In some cases, prior planning would have eliminated some of the delays encountered in determining the nature of the activity release.

In the nuclear incident at ICPP on Oct. 16, 1959, radiation alarms throughout the building started to sound at 0250, and all persons within the process building evacuated.[28] Within 45 min of the evacuation, a group of five people reentered the plant briefly to shut down the process that was still in operation. A RaLa run (a process involving the separation of radiobarium from short-cooled MTR fuel elements) had been completed on Oct. 15, 1959, and the spread of contamination was thought to be the result of a release of fission-product iodine from the process vent system. Emergency field equipment was set up by 0800 October 16. Contaminated-clothing and body-fluid samples were collected for radiochemical analyses. Radiation levels had decreased considerably by this time, but the RaLa equipment still seemed to be the cause of contamination. The first

indication that the incident might have been of nuclear origin was the report of the finding of Sr^{91} (9.7-hr half life) received from the radiochemical laboratory at about 1000 (7 hr after the alarms sounded). Substantial verification came an hour or so later when Ba^{139} (85-min half life) was identified.

The delay in identifying the nature of the incident was partly due to the fact that the chemical operations included high-level radioactive material. Previous releases of I^{131} had set off the sensitive air-monitor alarms. Consequently the RaLa equipment was first suspected. In addition, the criticality occurred on an off shift when there were few trained personnel immediately available.

On the other hand, the accidental excursion at the Oak Ridge Y-12 Plant on June 16, 1958, occurred in a fuel-processing area where no radioactive material other than uranium was being processed,[29,30] and hence an accidental criticality was suspected at once. Emergency planning could confidently presume that any activity release (airborne beta-gamma and high gamma readings) would indicate a nuclear excursion. The incident occurred during the normal work week, and thus qualified personnel were on hand to interpret the findings, which gave early evidence to the nature of the activity release.

This incident was of such a nature that the blue glow of Cerenkov radiation was visible to the operators in the immediate vicinity of the drum in which the excursion occurred. Further, an air filter trapped 18-min Rb^{88} and 32-min Cs^{138}, which were identified by gamma spectrometry. As a consequence of the rapid identification of the cause of the release, approximately 1200 persons were surveyed, and 12 exposed individuals were sent to the dispensary within 2 hr after the incident.

Radiochemical analyses were performed to determine the fission densities of the solution. These involved determinations of nine individual nuclides in order to obtain as reliable an estimate of the fissions per milliliter as possible. Many nondestructive analyses were performed soon after the incident in order to establish the levels of induced activities in the vicinity of the excursion. Neutron activation of the steel drum produced Co^{58} and Cr^{51} from $Ni^{58}(n,p)Co^{58}$ and $Cr^{50}(n,\gamma)Cr^{51}$ reactions, respectively. The radiochemical determination of these two nuclides in stainless-steel specimens gave an indication of the relative fast- and thermal-neutron fluxes at different positions within the drum. Radiochemical determination of Na^{24} in the blood of the exposed individuals was used as a means of determining the neutron dose to which the individuals were exposed (after suitable calibrations had been made using a burro and a mockup reactor).

In the plutonium release from the ORNL Thorex Pilot Plant,[31] cleanup operations were impeded because there was some delay in the correct identification of the problem. Radiochemical determinations and alpha spectrometry established the nature of the radioactivity release to be Pu^{239}.

In the activity release to the waste system and White Oak Creek at ORNL between Oct. 28 and Nov. 20, 1959, difficulty was encountered in locating the source of contamination.[32] After some delay in collecting samples, radiochemical analyses of samples from the waste system showed that the nuclides present matched those of the material being processed in Building 3019 and that some previous gross activity determinations had been in error.

On Nov. 11 and 12, 1959, contamination released to the environment from the gaseous waste-disposal system at ORNL[33] was identified by gamma spectrometry and radiochemical determinations as being due to radioruthenium.

In the ORNL incident[34] of April 1960, the nature and extent of contamination were determined in an area where there was the likelihood of previous low-level contamination. By means of gamma spectroscopy, it was determined that some activity found on a roof did not arise from the incident in question but that it was old contamination that had been deposited during the plutonium release from the ORNL Thorex Pilot Plant.

The recent incident at the SL-1 plant[35] at the National Reactor Testing Station (NRTS) was finally identified as being of nuclear origin by determining that Au^{198} had been induced in the ring of one of the employees involved. In addition, other induced activities such as Cu^{64} were detected in samples recovered from the accident. The presence of gross fission-product contamination on virtually all samples recovered from within the reactor building made the analysis for specific nuclides extremely difficult and delayed confirmation of a nuclear excursion.

These examples indicate the extent to which radiochemical determinations are valuable in nuclear emergencies or incidents. Because of the importance of early identification of the nature of the contaminant, it is advisable that emergency-planning procedures include the establishment of a central point at which emergency samples can be submitted and analyzed. It is likely that most of these analyses will have to be performed in a radiochemical laboratory, and the responsible personnel in the radiochemical laboratory should be on the emergency call list. In many instances these people are able to assist or advise in the judicious sampling procedures necessary to collect valid samples that will give information as to the nature of the incident in the shortest practical time.

The techniques of gamma-ray spectrometry, as described by Crouthamel[36] and Heath,[37] are a very valuable means of identifying gamma-emitting nuclides. The presence of short-lived fission products in a sample (as was the case in the Sr^{91} and Ba^{139} in the ICPP incident[28]) can be determined quickly and efficiently in most instances by these techniques. Even when a "background" of long-lived activity exists, peaks due to short-lived components can be detected. The development of solid-state particle detectors, such as those described by Blankenship and Borkowski,[38] has made the techniques of alpha spectrometry more reliable than in the past because of the greater beta tolerance and resolution of the solid-state detectors, as compared with Frisch grid instruments.

After the above-described instrumental techniques have been applied, specific radiochemical analysis will usually be needed. The choice of the analysis will depend upon the nature of the release. In any incident involving "old" fission products, the determination of Sr^{89} and Sr^{90} will be of paramount importance.

In summary, analytical radiochemistry should be an integral part of the emergency procedures for nuclear incidents. It is the radiochemist to whom the operations and health physics personnel usually turn for assistance in the determination or confirmation of the nature of a nuclear incident. A smooth and rapid line of communication should exist between the points of sample collection and analysis so that proper information can be transmitted to the operations groups in order to forestall any further damage to lives and property caused by the activity release. (J. S. Eldridge and S. A. Reynolds)

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Section V

CONSEQUENCES OF ACTIVITY RELEASE

Hazards Associated with Waste Treatment

Many papers have been written on radioactive waste processing and disposal, and the rate of publication seems to be increasing. The field seems to be well covered in three AEC bibliographies[1–3] covering 1598 technical reports and open literature papers published through May 1960. Of these publications, 187 were cataloged under "Radiological Hazards and Health Considerations," but they were concerned almost entirely with the hazards associated with environmental disposal and little, or not at all, with hazards associated with waste treatment prior to disposal. In papers on waste treatment, the hazards are usually assumed to be no greater than in first-cycle chemical processing of irradiated fuel. This seems to be reasonable, to a first approximation at least, since the fission-product activity levels are essentially the same and since, normally, the criticality hazard is eliminated by fissile-material recovery prior to waste treatment. Any particular waste-treatment plant should, however, be subject to the same safety analysis as that of a chemical processing plant with respect to the maximum credible accident and its containment. *Reactor Fuel Processing*[4] regularly reviews waste disposal and usually includes a review of safety in chemical processing. *Power Reactor Technology* recently reviewed the over-all waste problem from the economic point of view, with some comments on hazards.[5]

The present review is concerned primarily with recent papers on proposed processes for converting high-level radioactive wastes to solids prior to disposal and, especially, with the hazards aspects of these proposed processes. More than 30 papers were presented at the Second AEC Working Meeting on Fixation of Radioactivity in Stable Solid Media, held Sept. 27–29, 1960, at Idaho Falls, Idaho. Most of these papers were concerned with the high-temperature (400 to 1350°C) reduction of highly radioactive liquid wastes to solids, either directly or with the addition of glass-forming materials to fix the activity in less water-soluble form. The papers presented at this meeting are to be published in one volume[6] in the immediate future. (Some of the subjects were also covered elsewhere in prior publications.)

The fluidized-bed calcination process has received the most attention to date. Liquid waste is sprayed into a heated fluidized bed of granular waste solids and is thus essentially evaporated to dryness and calcined simultaneously in a process that, at steady state, is carried out under nearly isothermal conditions and with continuous removal of off-gases and solid product. Most of the work has been on waste of the aluminum nitrate type, and this work is now culminating in an engineering demonstration at ICPP.[7] The process is carried out at relatively low temperatures, 400 to 500°C, compared with other proposed calcination processes, and hence the solids may be more subject to the subsequent release of gases (e.g., nitrogen oxides from radiolytic and thermal decomposition of residual nitrates) during long-term storage of highly radioactive wastes.[8] This in turn probably means vented storage and surveillance to guard against hazardous atmospheric contamination. The use of air as the fluidizing gas gives large volumes of off-gas containing both entrained and volatile radioactivity that must be decontaminated before discharge.[9] The demonstration plant has a fairly elaborate off-gas treatment system, and no hazard to the public

is expected from activity in the off-gas discharged through the plant stack. Fluidized-bed calcination of Purex type waste is also under study.[10]

Rotary-ball-kiln calcination of aluminum nitrate wastes at temperatures up to 550°C and of aluminum nitrate wastes containing zirconium fluoride at temperatures up to 700°C has been studied extensively at Brookhaven National Laboratory (BNL).[11,12] A continuous steady-state process is used which produces a granular oxide powder similar to that produced by the fluidized bed. The product should be less subject to off-gassing during storage than that from the fluidized bed because of the higher temperatures used during calcination. The off-gas problem is much less severe than that with the Idaho demonstrational fluid-bed calciner, which was designed to use air as a fluidizing medium, and therefore generates a large amount of noncondensables. After condensation and scrubbing, the amount of off-gas to be treated and discharged from the rotary-ball-kiln calciner is small and is probably determined by air inleakage if the process equipment is operated at subatmospheric pressures. Maintenance of the calciner itself may offer more potential hazard to operating personnel than that with the fluidized bed; that is, the use of moving machinery at higher temperatures may necessitate more maintenance and hence more exposure of operating personnel. This problem should be solvable, however, by proper design and by the use of remote maintenance techniques. The apparent advantage of the rotary-ball-kiln process over the Idaho fluid-bed process, in respect to off-gas production and possible hazardous atmospheric contamination, would disappear if the latter were operated with either steam or recycle gas as the fluidizing medium instead of air. Use of air in the Idaho installation was a design choice dictated by consideration of limited capital funds and is not inherent in the fluidized-bed process.

Radiant-heat spray calcination of Purex type wastes at temperatures up to 800°C has been developed at Hanford.[13,14] This, too, is a continuous steady-state process that produces a granular powder. The higher temperature presumably gives a more stable product and, possibly, more maintenance problems, although there are no moving parts in the calciner proper. This process produces a low volume of the rotary-kiln product, but the spray calciner off-gas may require somewhat more decontamination since the finely divided solids are formed in the gas phase and at higher temperature. Ruthenium volatility must be dealt with in all nitric acid calcination processes, and at 800°C cesium volatility may also be a problem, although probably not in sulfate-containing wastes such as Hanford Purex wastes. Again, although the routine operation of high-temperature equipment may not be as trouble-free as low-temperature equipment used in conventional fuel processing, there seems to be no reason to expect an undue hazard to the public. An explosion, for example, is not considered likely, and containment of activity release from other equipment failures should be possible with standard chemical processing cell design.

Pot calcination is being developed at ORNL and at Hanford.[15,16] In one version of this process, the liquid waste is fed into a hot (900°C) stainless-steel pot at a rate slow enough that the water is driven off and the bulk of the nitrates decompose at the same rate as that at which they are fed. This type of operation is almost equivalent to continuous steady-state operation. It likewise produces a low volume of off-gas as does rotary-kiln and spray calcination, while being simpler mechanically and giving a more dense and more stable product. This type of pot calcination has, however, an inherently low production rate per pot because of heat-transfer limitations. For this reason most of the development work on pot calcination has been on semicontinuous unsteady-state operation; that is, the pot is filled with liquid waste initially, and additional feed is added to keep the pot full. The feed demand rate drops with time as water and nitric acid are evaporated and solids start to form on the walls of the pot. A point of diminishing return is reached, at which time feeding is discontinued but heating is continued to decompose nitrates and raise the solid residue to a final temperature approaching 900°C. This unsteady-state process gives reasonably high production rates, a good-quality solid residue, and low off-gas volumes. It is, however, probably potentially more hazardous than the low-holdup continuous steady-state processes since the pot contains appreciable amounts of water and/or nitrates during a large fraction of the calcination cycle, and the operation cannot be quickly shut down; even after shutting off feed and power, the system holds enough heat to con-

tinue generation of steam and/or nitrogen oxides for some time. Thus a plugged off-gas line, for example, might lead to excessive pressure buildup in the pot, followed by rupture of the pot or connecting lines. Another danger is that any excessive amounts of organic matter in the feed (e.g., entrained solvent and its degradation products) might accumulate in the pot to a point where a violent chemical reaction between the organic and the nitric acid and nitrate salts could rupture the equipment. The development groups recognize these potential hazards and think that they can be eliminated by proper design of equipment (e.g., by controlled venting of any excessive pressure surges). Laboratory experiments at ORNL have not been able to produce any violent reactions during the calcination of synthetic Purex waste containing entrained TBP-Amsco solvent.

The solids produced by the calcination processes mentioned above may be radiolytically and thermally stable enough for "ultimate" disposal, but they cannot be stored in such a manner that they might come in contact with ground water because their radioactivity is present in highly leachable form. Considerable attention has been paid in the United States and in other countries to incorporating fission-product wastes into glass to "fix" them in a stable, insoluble form. The Canadians have carried their batch process through a hot-cell demonstration, followed by a test burial in contact with ground water.[17,18] On heating to 1350°C to form the melt, substantial fractions of the ruthenium and cesium were volatilized, but they were mostly removed from the off-gases by adsorption on iron oxide and fire brick. It is believed that these materials could be recycled to subsequent glass batches. Solubility of the activity in the glass, although orders of magnitude lower than with nonglass calcines, has not been shown to be low enough to permit safe disposal in direct contact with ground water. It does seem fair to say, however, that incorporation of the fission products in glass amounts to an additional "barrier" between the activity and the environment.

The British[19] are planning a radioactive pilot-plant test of their process in 1961. This process includes addition of borax and silica to the liquid waste and feeding the slurry to a 900 to 1000°C pot in the relatively slow, almost continuous, almost steady-state manner mentioned above. As the solid level builds up in the bottom of the pot, the temperature is gradually raised from the bottom up to 1150°C to melt the mixture and form the glass. Because of the rather drastic nature of the evaporation stage, fine dust is carried over with the off-gas to an absolute filter that will be incorporated in a subsequent batch of glass. The stainless-steel pots thus become the ultimate container of the solidified glass. Cesium volatility has been found to be less than 0.1 per cent at temperatures below 1150°C, but 10 per cent ruthenium volatility is expected. The ruthenium is to be adsorbed on silica gel or iron oxide that will be incorporated in the subsequent batch of glass.

The glass processes give the safest type of solid for ultimate disposal, although the glass-making operation itself may be somewhat more hazardous than the lower temperature calcination processes. There does not seem to be any inherent hazard that cannot be safely designed for. (J. T. Roberts)

Use of Routine Meteorological Observations for Estimating Atmospheric Dispersion

By F. A. GIFFORD, Jr.

Estimates of atmospheric dispersion are essential information in the selection of a reactor site and in the evaluation of the hazards of reactor operation. In selecting a site, the dispersion characteristics of the atmosphere at the various sites under consideration are important because most reactors, if not all, generate or induce some atmospheric radioactivity during routine operation and because there is the possibility of accidental release of radioactivity to the atmosphere. Since few reactor installations have permanent meteorology staffs, the burden of providing dispersion estimates ordinarily falls on health physics or reactor operations personnel. Moreover, it can be expected that the national Weather Bureau forecasting offices near reactor sites will be increasingly called upon to provide technical advice in this area. At present, only a few forecasters are familiar with low-level dispersion problems, and consequently it is desirable that simple, easily applied methods of estimating atmospheric dispersion, preferably those employing routine rather than highly

specialized meteorological observations, be made available. One suggestion for accomplishing this was described by F. Pasquill in an unpublished note, the substance of which has recently been presented in papers by Meade.[20,21]

Pasquill and Meade define a plume height, H (in meters), and an angular plume width, ϕ (in radians), such that the concentration at the edge of the plume equals 10 per cent of its value on the plume axis. It appears (cf. Gifford[22]) that the visible edge of a smoke plume is fairly well approximated by this 10 per cent value. Pasquill and Meade estimated values of H and ϕ as functions of distance from the source for a range of meteorological conditions, using reasonable meteorological principles and incorporating existing observations of dispersion. (A summary of recent dispersion observations has been presented in *Nuclear Safety*.[23])

The quantities H and ϕ are directly related to the particular values of $2y_0$ and z_0, defined in plume width and height formulas given previously (Gifford,[24] Eqs. 10 and 11), for the special 10 per cent case. Consequently, H and ϕ can be expressed in terms of the dispersion coefficients, σ_y and σ_z, of the generalized Gaussian plume model,[24] by the relations

$$H = 2.15\sigma_z \qquad (1)$$

and

$$\tan\frac{\phi}{2} = 2.15\,\frac{\sigma_y}{x} \qquad (2)$$

where the numerical coefficient 2.15 is the 10 per cent ordinate of the normal distribution curve. Use of σ_y and σ_z, rather than ϕ and H, seems desirable in view of their interpretations as standard deviations of the plume concentration distribution and of their increasing widespread use in summarizing atmospheric dispersion data.[23]

Families of curves of σ_y and σ_z (in meters) as functions of various meteorological categories are presented in Figs. V-1 and V-2. These curves are based on the values of H and ϕ given by Pasquill and Meade, as converted by Eqs. 1 and 2. The manner of relating these curves to prevailing conditions of wind speed and to factors influencing atmospheric stability is evident from the data given in Table V-1, which is based on the paper by Meade.[20]

Observations of the independent parameters given in Table V-1, namely, wind speed and cloudiness, are commonly carried out at most meteorological installations and, in any case, present no great instrumentation problems. In practice, values of σ_y and σ_z would be estimated from Table V-1 and Figs. V-1 and V-2, and ground-level air concentration values, χ (in grams or curies per cubic meter), would

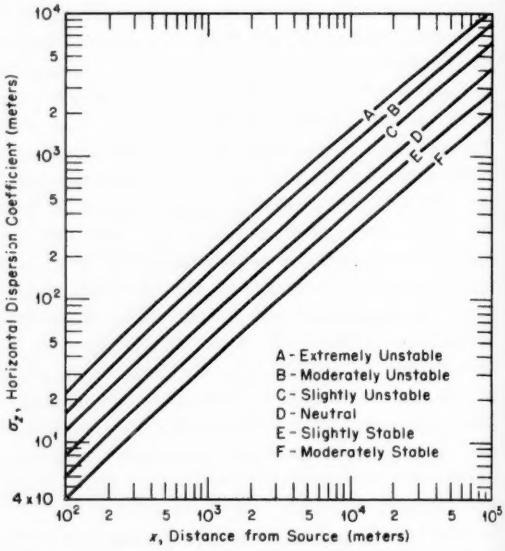

Fig. V-1 Horizontal dispersion coefficient as a function of distance from source.

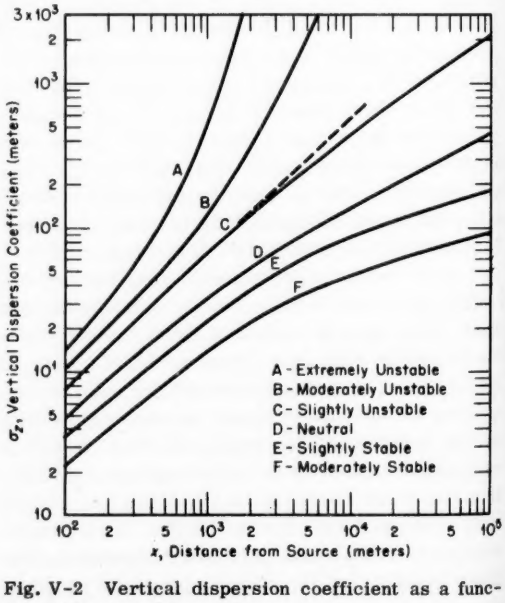

Fig. V-2 Vertical dispersion coefficient as a function of distance from source.

Table V-1 METEOROLOGICAL CATEGORIES

A: Extremely unstable conditions
B: Moderately unstable conditions
C: Slightly unstable conditions
D: Neutral conditions*
E: Slightly stable conditions
F: Moderately stable conditions

Surface wind speed, m/sec	Daytime insolation			Nighttime conditions	
	Strong	Moderate	Slight	Thin overcast or ≥4/8 cloudiness†	≤3/8 cloudiness
<2	A	A-B	B		
2	A-B	B	C	E	F
4	B	B-C	C	D	E
6	C	C-D	D	D	D
>6	C	D	D	D	D

*Applicable to heavy overcast, day or night.

†The degree of cloudiness is defined as that fraction of the sky above the local apparent horizon which is covered by clouds. [*Manual of Surface Observations* (WBAN), Circular N(7th ed.), paragraph 1210, U. S. Government Printing Office, Washington, July 1960.]

then be computed, using the generalized Gaussian plume formula:

$$\chi = \frac{Q}{\pi\sigma_y\sigma_z u} \exp\left[-\frac{1}{2}\left(\frac{y^2}{\sigma_y^2} + \frac{h^2}{\sigma_z^2}\right)\right] \qquad (3)$$

where

Q = source strength in grams or curies per second
u = average wind speed in meters per second
h = height of the source above the ground in meters
y = lateral (crosswind) distance from the plume axis in meters

The right-hand side of Eq. 3 contains a multiplicative factor of 2, which is a conventional device to account for the assumed reflection of the plume by the ground plane.

Meade[20] suggests several important modifications of these estimated dispersion coefficients to take into account various real complications that may be encountered in practical applications. The following points are especially significant. (The following excerpts* have been taken directly from Meade's paper.[20])

Estimation of Vertical Spread of Cloud or Plume

As the distance from the source increases, the cloud or plume extends higher into the atmosphere. The rate at which this vertical spreading takes place depends mainly on the degree of stability of the atmosphere in the layer concerned. The most rapid spreading occurs in very unstable conditions, the slowest when the atmosphere has great stability, as in the case of a sharp inversion.

In applying Fig. 2 [see Fig. V-2*] to the conditions in the atmosphere on any occasion, it is important to consider a deep layer in case any discontinuities of temperature gradient are present. Quite frequently it may be found that, while the atmospheric layer adjacent to the earth's surface is unstable with strong convection and mechanical turbulence taking place, the next layer above is an inversion, and this would act as a barrier to further vertical spreading of the plume.

In the event of a strong inversion "lid" at some height H, the value of σ_z should be considered constant at distances beyond the point where

$$\sigma_z = \frac{H}{2.15} \cong \frac{H}{2}$$

In very light winds (<2 m/sec) on a clear night, i.e., conditions productive of sharp ground frost or heavy dew, the vertical spread will be even less than the values given for category F in Fig. 2 [see Fig. V-2]. No estimates are given for such a case because, in practice, the plume from a ground source is unlikely to have any definable travel.

*Reprinted here by permission from the Comitato Nazionale per le Ricerche Nucleari, Rome.[20]

*Figure 2 of reference 20 is the same as Fig. V-2 of this review, except that the abscissa of Fig. 2, reference 20, was expressed in terms of H rather than in σ_z.

Estimation of Lateral Spread of Cloud or Plume

The ϕ values given by Meade, and consequently the σ_y values in Fig. V-1, are estimated to apply to air concentration values, χ, averaged over a few minutes' time. When the wind direction is variable, or when longer period concentration averages (an hour or more) are involved, it is suggested that values of lateral spread be based on the wind-direction range, that is, the difference between the extreme values of wind direction during the period. The rules suggested by Meade can conveniently be expressed in terms of Fig. V-1, curve A, for extremely unstable conditions, as follows:

1. When the wind direction is variable, the value of σ_y given by curve A, Fig. V-1, should be multiplied by the wind-direction range (in degrees) measured over a period of time equal to that during which the concentration is to be averaged.

2. For concentrations averaged over periods of 1 hr or longer and for distances greater than a few kilometers, the range of 15-min average values of wind direction, rather than instantaneous values, should be used.

A more technical discussion of the effect of short-time-period averaging on air concentration values can be found in a recent paper by the writer,[25] which moreover summarizes the currently available observations on this subject.

Estimation of Wind Speed and Direction[20]

The choice of wind values is governed mainly by the distance from the source and then by the estimated value of *H*, the vertical spreading. For the shortest distance, 0.1 km, the surface wind speed as measured by the anemometer should be used, while for a distance of 100 km an estimate should be made of the mean wind through a layer of height *H*. In practice, for a vertical spread from a few hundred metres, the vector average of the surface and geostropic winds, backed by 10°, should give a reasonably good estimate for a distance of 100 km.

With the winds for 0.1 km and 100 km determined, those for the intermediate distances 1 km and 10 km are obtained by interpolation at equal intervals.

In arriving at these estimates, attention should be given to the period of release. For a long release the basic estimates of wind should be made for the total period of release, taking account of the distance travelled. If the wind varies with distance, appropriate allowance should be made with the aid of synoptic charts. Even in the case of a short release, the important factor may be the trajectory over a long distance, and this is again best obtained by an average wind corrected as necessary for spatial variations.

Somewhat more refined estimates of the wind may be made by computing a true mean through a layer of depth *H*, weighting the wind at the various levels according to concentrations.

When Eq. 3 and the various special equations for maximum concentration, fumigation, rainout, etc.,[24] are used in connection with the generalized Gaussian plume formula, the dispersion values suggested by Pasquill and Meade (as expressed in terms of σ_y and σ_z in Figs. V-1 and V-2) provide a straightforward approach to the problem of estimating air concentrations downwind from an isolated source. It should be emphasized that the dispersion parameter values proposed by Pasquill and Meade were presented solely as the best estimates currently available. The curves of Figs. V-1 and V-2 will, in all likelihood, have to be modified as air concentration measurements, particularly at the greater distances, become available. A recent series of diffusion observations by Kawabata[26] has, however, verified the above method out to distances of 5 km from the source.

Nevertheless, use of these dispersion estimates in connection with reactor safety studies, as well as other air-pollution problems, particularly in the form of the plume standard deviations presented here, seems to be very desirable. It is a straightforward approach that directly reflects the best current observational knowledge of air concentrations. Moreover, adjustments to Figs. V-1 and V-2 and to Table V-1 can easily be made as more observational data are acquired. In all likelihood, major adjustments will never be necessary. Furthermore, results of improved theoretical understanding of atmospheric dispersion, such as Hay and Pasquill's recently proposed "method of moving averages,"[27] or the writer's "fluctuating plume model,"[28] can be readily incorporated into the scheme via Eq. 3, the generalized Gaussian plume formula. Likewise, interpretation in terms of older methods, such as Sutton's well-known dispersion formula, is straightforward, as has previously been pointed out.[24] The generalized Gaussian dispersion formula is, briefly, both adjustable and compatible,

and thus provides an appropriate framework into which to fit existing atmospheric dispersion knowledge, both theoretical and observational, as well as anticipated future improvements.

(F. A. Gifford, Jr.)

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Section VI

CURRENT EVENTS

Redox Multipurpose Dissolver Incident

A combination of fires and explosions or sudden pressurizations occurred in a uranium metal dissolver in the Redox plant at Hanford on Apr. 17 and 18, 1960. There was no exposure of operating personnel to radiation and no significant release of activity to the building or plant environs. The immediate consequences of the accident were the loss of the dissolver and the contamination of the dissolver cell. The incident is discussed in a classified report[1] that thoroughly analyzes the causes and makes specific recommendations for avoiding similar occurrences in the future.

The incident took place in cell B of the Redox canyon in a new multipurpose dissolver[2] (see Fig. VI-1). The dissolver contained dejacketed and partially dissolved uranium metal slugs that were half covered with water and had not been disturbed since 11:40 p.m. on April 15. The first indication of trouble (recognized after the fact) was a gradual increase in the temperature of the gases leaving the dissolver off-gas condenser, beginning at 10:00 a.m. on April 17. The temperature of the water in the dissolver began to increase at noon and had increased from 60 to 86°C by 2:30 p.m. At 2:30 p.m. a small amount of water was added to the dissolver when an operator attempted to reestablish a water seal on the dissolver lid, and a temperature rise occurred at an adsorber bed in the dissolver off-gas system (probably from hydrogen in the off-gas).

The water in the dissolver was cooled with an internal cooling coil, and, after frequent pressurizations, the dissolver vacuum was finally reestablished. At 6:45 p.m. the water was drained from the dissolver, and the slugs were

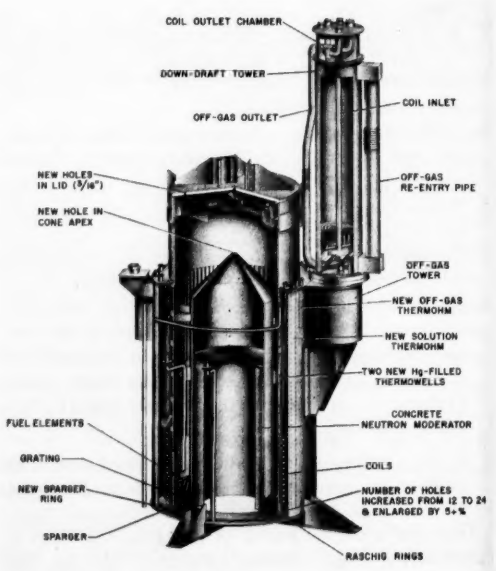

Fig. VI-1 Sketch of Redox multipurpose dissolver.

half covered with 47 per cent nitric acid. Almost immediately the specific gravity rose to greater than 1.85, indicating a solution of more than 2.5*M* uranium. The dissolver pressurized frequently after the acid addition, and the vacuum was not reestablished until 7:40 p.m.

The dissolver was then heated to boiling, and a small amount of 60 per cent HNO_3 was added. The vessel vacuum became almost uncontrollable, and therefore the acid addition was stopped and cooling-water flow was started. The liquid level in the dissolver increased, indicating a leak in the cooling coil. Operation became more erratic, and a rumbling was heard in the cell at 8:45 p.m. The coil cooling-water flow rate was increased to full capacity, and the

liquid-level rise increased in rate. The second and third rumbles were heard at 15-min intervals, and, at the end of that time, the liquid level in the dissolver was twice its original value. Cell off-gas samples taken during the 2-hr period immediately preceding the rumbles showed that airborne gamma radioactivity increased to 300 times normal from 6:30 to 8:30 p.m.

The cooling-coil water was then turned off; the dissolver liquid level dropped; and the cell sump level rose. Samples taken during the next shift confirmed that a large amount of solution containing approximately 0.3*M* uranium was on the cell floor.

At 1:30 a.m. the dissolver cell cover was removed, and the interior was inspected with remote-viewing equipment. It was found that the dissolver lid was inverted and was 1 to 2 ft off center. There was no other visible damage, and, although the underside of the lid was discolored, the remainder of the cell interior appeared to be clean. The cell cover was then replaced.

Beginning at 4:00 a.m. on April 18 and continuing throughout the day, the temperature of the gases in the dissolver off-gas condenser rose slowly, and the gamma activity in the cell exhaust increased. By 7:30 p.m. it was apparent that the reaction in the dissolver was starting again, and therefore a large quantity of water was added as rapidly as possible to the dissolver. The pressure in the dissolver increased (even though the lid was partly open), and the off-gas temperature rose rapidly and then decreased slowly over the next 6 hr. The cell exhaust gamma activity peaked at 100 times normal and then returned to normal by 10:00 p.m.

At no time during the first or second reactions did a hazardous quantity of radioactivity pass the vessel or cell exhaust filters. To prevent further incidents, the dissolver was purged continuously with an inert gas, and the contents were "drowned" with water every 8 to 12 hr until most of the uranium had been removed in the cleanup process.

At 10:00 a.m. on April 19, the cell cover was removed a second time to replace the dissolver lid in its proper position. Air activity in the canyon increased, and the attempt to reinstall the dissolver lid was unsuccessful. While the cell cover was off, it was observed that fine dust covered everything in the cell. The cell was closed and was not reopened until the problem of the spread of activity from the cell could be solved.

Cleanup

The first step in cleanup was to install a backup filter downstream from the dissolver off-gas filter to minimize the hazard during dissolver emptying and decontamination. At the same time, cleanup was started to remove the surface contamination in the canyon and to decontaminate the crane. This was done by spraying and flushing with water and, in the case of the crane, with solvent. Before the dissolver cell was reopened, several thousand gallons of water were sprayed into the cell with fog nozzles on flexible metal hoses inserted through existing pipes through concrete. A water spray ring was installed around the cell top to provide an umbrella of water above the cell opening. Then the cell roof blocks were removed without incident.

The old dissolver lid was replaced with a new lid containing a vapor-and-liquid temperature-sensing element and a water-addition line. A nozzle blank sample was removed for neutron-activation detection. Then the floor of the cell was leached with 20 per cent nitric acid to remove soluble sludge.

The remaining uranium was removed in two steps. First, the reactive uranium (probably present as oxide) was leached from the dissolver with dilute nitric acid. The dissolution was rapid, and stoichiometric calculations indicated that UO_2 or U_3O_8 was being dissolved. In the second step, a stronger acid was used to remove the remaining metal. During this phase of the cleanup, dissolver temperatures rose very rapidly on two or three occasions until the reaction was quenched by flooding the dissolver with water. The presence of hydrogen on each occasion, as detected by an off-gas analyzer, suggested that the remaining metal had caught fire again. Nevertheless, the dissolver was finally emptied in this manner.

The interior of the dissolver was then given several chemical decontamination flushes, after which the cell and the dissolver interior and exterior were inspected with the use of a remote television camera. Two holes could be seen inside the dissolver, and there was some irregularly shaped material that resembled

welding slag on the bottom of the dissolver. After remote inspection, the dissolver was removed to a decontamination facility. Closer inspection revealed that the holes were probably caused by high temperatures rather than pressures. An irregular slit was also found in the top of the cooling coil.

Chemical analysis of the slag from the dissolver disclosed that the sample was 72 per cent iron, 9 per cent chromium, 11 per cent nickel, and 8 per cent uranium. Phase diagrams of the four-component system indicated that temperatures in excess of 1300°C must have been present in the dissolver during the incident.

Causes of the Accident

The incident was probably the result of chemical reactions. Criticality was ruled out early as a cause by (1) calculating the amount of energy necessary to raise the temperature of the dissolver contents as shown on the instrument charts, (2) estimating the number of fissions required to produce that energy, and (3) estimating the attendant increase in gamma-activity background in the cell. The actual change in background was very much smaller than the calculated value. The conclusive evidence that it was not a criticality accident was furnished by the nozzle blank sample, which was removed from the cell for neutron-activation analysis. No activation was found.

The most probable chemical cause of the incident is described below. After dejacketing and partial dissolution, the remaining uranium had a high ratio of surface area to volume, and the surface was very active. The uranium was only partly covered with water and then stored for more than 30 hr. The surface of the metal oxidized, and the temperature of the metal slowly rose from the oxidation reaction and from fission-product decay heat.

The new dissolver was designed so that the metal packing was closer than in previous dissolvers. Consequently the rate of heat dissipation by natural convection to air in the unwetted portions of the material was reduced. The metal temperature rose slowly to approximately 300°C, where the metal oxidation rate is rapid and is controlled by gas diffusion through the oxide scale to the surface. Wherever the scale was disrupted, ignition occurred, and temperature excursions to at least 1300°C took place. Uranium metal melts at 1100°C.

The oxide material formed by the slow and rapid oxidation reactions, part of which was probably collected on the bottom of the dissolver, accounted for the initial very fast dissolution rate in dilute acid to greater than 2.5*M* uranium. The molten uranium reacted with the stainless-steel dissolver walls and the cooling coil to produce holes and slag.

The cooling coil failed at approximately the time of addition of the 60 per cent acid. In an attempt to remove heat from the dissolver, the cooling-water flow rate was increased, and the uranium was flooded. The rumbles were probably the result of cold water contacting hot, or even molten, uranium metal and producing steam or hydrogen, which may have ignited explosively with oxygen or nitrogen oxides. After the third rumble, the cooling water was turned off, and it drained from the dissolver, uncovering half the uranium.

The metal required only 8 hr to reignite, since it was probably activated by contact with the water (because of hydride formation or removal of scale). The second fire was quenched after 15 hr, and this quenching reaction was sufficiently violent to spread oxide particles throughout the cell. These particles were carried out of the cell during the second inspection, and they contaminated the canyon and the crane.

Thus the interrupted dissolution and incomplete coverage of the remaining metal were the prime causes of the incident, but fuel geometry and heat-transfer problems necessarily were also present. Inadequate instrumentation also contributed to the incident, since vapor temperatures, dissolver wall temperatures, and off-gas hydrogen-content values were not available with which to assess the extent of the problem. However, although the temperature of the gas entering the dissolver tower (which would have been invaluable) was not available, it seems highly probable that ignition could have been prevented had the rise of temperature of the gas leaving the tower been properly observed and heeded.

There was a series of questionable reactions to events which may also have contributed to the incident. The decision to continue with the dissolution in view of unexplained rising temperatures may have been a poor one in retrospect. However, it should be pointed out that dissolution and removal of the burning uranium may have been the only safe way to proceed

once the fire began, since it could allow controllable, low-temperature oxidation of the metal. Further, the explosions apparently did no significant damage to equipment. The "instantaneous" dissolution after the weak acid addition was not properly interpreted; it could have suggested the presence of large quantities of uranium "ash." The delay of 15 hr in drowning the second fire may have increased the violence of the final pressurization; this pressurization did result in severe contamination problems.

There were extenuating circumstances surrounding the causes of the incident, as well as possible secondary factors that may have contributed to it. The acceptance of incomplete water coverage of dejacketed slugs was based on years of trouble-free operation. The unimportant temperature increases, always observed when dissolver operation was temporarily interrupted, were attributed principally to fission-product decay heating. The possible effect on heat dissipation of a change in slug packing in the new dissolver was not stressed; the detailed analysis of the effect of temperature on the rates of heat production and dissipation in the new dissolver was carried out after the incident.

Conclusions and Recommendations

It was concluded that the dissolver incident was caused by autooxidation of dejacketed, partially dissolved uranium, followed by steam pressurizations or hydrogen explosions. Essentially all the internal damage to the dissolver was the result of the alloying of stainless steel and uranium. The ignition of the metal resulted from the closer slug packing in a new dissolver, which changed and made unstable the heat production and dissipation characteristics of the slugs that were not covered with water.

It was recommended that dejacketed metal in dissolvers be covered with water at all times, if possible, and be uncovered less than 8 hr in any case. Improved dissolver instrumentation was also recommended so that, for example, the dissolver vapor temperature could be monitored. Although there was no spread of radioactivity to the occupied zones of the building or to the plant environs, it was recommended that backup filters be installed in the vessel off-gas lines to minimize the hazard of airborne particulates. (J. C. Bresee)

Accidents in Nuclear Energy Operations

Incidents that occur in nuclear energy operations are of particular interest to everyone concerned with nuclear safety, and, as detailed information on such incidents becomes available, it is discussed in *Nuclear Safety*. The significant accidents that occurred from 1945 through 1958 have been summarized in an AEC report[3] and its supplements. Another AEC report[4] contains a compilation of radiation incidents involving materials in transit. In addition to these reports, D. F. Hayes, Chief of the Safety and Fire Protection Branch, AEC Office of Health and Safety, has recently released three compilations of recent accidents in AEC and licensee operations.

The three accident compilations include: (1) serious accidents in AEC plants from Jan. 1 to June 30, 1960, (2) incidents involving radiation at licensees' plants from July 1, 1958, to June 30, 1960, and (3) incidents involving radiation in AEC operations from July 1, 1958, to June 30, 1960. These lists are published in the AEC Annual Report to Congress for 1960.[5] The first compilation includes all incidents at AEC-owned plants, including those not involving radiation but which resulted in injuries and property loss. There were 22 such accidents involving four persons killed and three persons injured, as well as a loss of over a million dollars. The second compilation includes all incidents in licensee-owned facilities which are required to be reported by the *Code of Federal Regulations*.[6] The list includes 36 incidents involving 67 persons. The third compilation lists 22 incidents involving radiation in AEC operations, including one fatality.

All these incidents are of interest to operators of nuclear energy facilities to the extent that they may reveal weaknesses in the design of equipment or the control of processes. Studies of such weakness might indicate means for preventing the occurrence of similar instances in other installations. Table VI-1 presents a chronological compilation of all accidents that directly involved nuclear equipment or activity release. Those incidents which have been reviewed in *Nuclear Safety* are so indicated. Other incidents of interest will be reviewed as the detailed information is received. Inasmuch as the first issue of *Nuclear Safety* did not appear

(Text continues on page 63.)

Table VI-1 CHRONOLOGICAL LISTING OF ACCIDENTS IN NUCLEAR ENERGY OPERATIONS

Date	Location	Nuclear Safety Review Citation	Installation Owner	Nature of Incident	Extent of Exposure
August 1958	Vallejo, Cal.		Licensee	Engaged in radiography work	One person received a film badge exposure of 4.8 rem during month
August 5–6, 1958	Portsmouth, N.H.		Licensee	Mechanical failure of a camera containing a 7-curie Ir^{192} source	Four persons received film badge exposures ranging from 0.6 to 7.5 rem
September 5, 1958	St. Louis, Mo.		Licensee	5 to 10 millicuries of P^{32} boiled over onto a hot plate from an oven container; in a second incident, about 30 min later, a centrifuge tube containing ~100 millicuries of end product turned over in the water bath and resulted in a loss of 2.58 millicuries of P^{32}	None
September 15–29, 1958	Ithaca, N.Y.		Licensee	Shielding removed from a source in the mistaken belief that 15 curies of Co^{60} had been previously removed	One person received a calculated exposure of 10 rem
September 22, 1958	Houston, Texas		Licensee	Over exposures occurred when the employees were retaping the extension cable of a radiographic device	Film badge exposures of two persons were 3605 and 1500 mr
October 28, 1958	Burlington, Mass.		Licensee	Employee accidentally pushed protruding source back into its container with his bare hand	One person received a film-badge exposure of 1.4 rem whole body and a calculated 100- to 700-rem exposure to tips of fingers of right hand
October 28, 1958	Cambridge, Mass.		Licensee	5 millicuries of Po^{210} solution was spilled when its glass container exploded while being opened	Negligible
October 29, 1958	Arlington, Mass.		Licensee	Malfunction of equipment resulted in failure of a 400-curie Ir^{192} source to return to shielded position	One person received a film-badge exposure of 1.435 rem whole body and a calculated hand exposure of 1000 rem
November 6, 1958	Boston, Mass.		Licensee	During the transfer of several hundred curies of tritium from a shipping container to a vacuum manifold, a valve was inadvertently left open, and 200 curies of tritium was lost	Calculations showed that the average concentration of tritium in the effluent discharged through the stack during a 24-hr period amounted to 6.6×10^{-13} μc/ml

Table VI-1 (continued)

Date	Location	Nuclear Safety Review Citation	Installation Owner	Nature of Incident	Extent of Exposure
November 12, 1958	Bronx, N.Y.		Licensee	A leak occurred in a 1000-curie Co^{60} sealed source contained in a teletherapy unit	None
November 14, 1958	Houston, Texas		Licensee	A radioactive sealed source was found at the head of a filter casing near a pump on a pipeline	Eight persons received radiation doses calculated to be from 0.13 to 1 rem whole body and 4 to 560 rem to the hand
November 17, 1958	Pittsburgh, Pa.		Licensee	Exposure probably caused by failure to close and lock camera between shots while using 17-curie irridium source	One person received a film-badge exposure of 10 mr for week beginning Nov. 15, 1958
November 21, 1958	Tonawanda, N.Y.		Licensee	Leaking source caused contamination of equipment in a hot cell	None
December 3, 1958	Cambridge, Mass.		Licensee	A radiographer neglected to crank the source back into the camera and walked into the radiation area where the 52-curie Ir^{192} source was exposed	One person received a film-badge exposure of 6 rem whole body
December 4, 1958	Miamisburg, Ohio		AEC	An employee inhaled Po^{210} during the handling and counting of a polonium alpha source	One person received an estimated 3.5 times the maximum continuous body burden for Po^{210}
December 30, 1958	Los Alamos, N.M.		AEC	Plutonium suspended in an emulsion in a 225-gal tank assumed a critical configuration (results of this accident have been the subject of several technical and scientific reports)	One person died 35 hr later from the effects of a radiation dose estimated to be several thousand rems to the midportion of the body; two others received whole-body radiation doses of up to 118 rem
January 6, 1959	Livermore, Cal.		AEC	Malfunction of remote controls on an electron accelerator caused exposure to personnel	One person received an exposure dose of 41 rem; another received an exposure dose of 400 rem; five others received less than 50 mrem
February 4, 1959	Shrewsbury, Mass.		Licensee	Approximately 35 millicuries of C^{14} lost when a beaker containing 80 millicuries of sodium acetate in alcohol suddenly boiled over	Negligible
February 9, 1959	Waltham, Mass.		Licensee	Flask shattered containing ~50 millicuries of C^{14}	None

Table VI-1 (continued)

Date	Location	Nuclear Safety Review Citation	Installation Owner	Nature of Incident	Extent of Exposure
March 2, 1959	Albany, N.Y.		Licensee	A mold fractured and allowed molten normal uranium to flow into a mold tank and onto the floor of the furnace area	None
March 19, 1959	Durango, Cal.		Licensee	A fire occurred in a uranium refinery caused by a momentarily unattended open flame used for drying the final mill product	None
March 31, 1959	New York, N.Y.		Licensee	Spill of 0.1 millicurie of P^{32} as a result of filling centrifuge tubes too full	Calculated upper limit of exposure to ~300 cm^2 of skin of one person was ~1.50 rad
March 31, 1959	Richland, Wash.		AEC	An explosion occurred inside a glove box while plutonium was being machined	Negligible
April 1–December 31, 1959	Spokane, Wash.		Licensee	Employees were exposed during diagnostic and therapy work with I^{131} and work with x-rays	One person received a quarterly film-badge exposure of 5150 mrem; another person received film-badge exposures of 7000 mrem, 3400 mrem, 2900 mrem during three successive quarterly periods
April 2, 1959	Aiken, S.C.		AEC	A small volume of contaminated solvent leaked while being transferred to storage and dripped onto the plant roadway; cleanup cost, $8700	None
April 17, 1959	Bloomsburg, Pa.		Licensee	Tritium gas escaped while being transferred from defective tubes into closed containers	Calculated whole body dose of ~9 rem; urine was found to contain 668 μc of tritium/liter
May 5, 1959	Hempstead, N.Y.		Licensee	An employee inadvertently spilled between 100 and 200 μc of I^{131}	Negligible
July 3, 1959	Berkeley, Cal.	Vol. 1, No. 4, p. 88	AEC	Overpressurization of a box for helium cooling at an accelerator installation blew out a thin experimental foil containing about 0.05 curies of Cm^{244}; cleaning and lost time cost, $58,500	None
July 15, 1959	Los Alamos, N.M.		AEC	A unit filtering system caught fire from sparks or hot metal from a welding torch; alpha-contamination cleanup cost, $2000	None

Table VI-1 (continued)

Date	Location	Nuclear Safety Review Citation	Installation Owner	Nature of Incident	Extent of Exposure
July 17, 1959	Los Alamos, N.M.		AEC	$10,300 of tritium was lost when the gas was inadvertently released through the exhaust system	Negligible
July 23, 1959	Syracuse, N.Y.		Licensee	A process loss of ~25 curies of Kr^{85} occurred by exhaust to the stack	None
August 6, 1959	Miamisburg, Ohio		AEC	A spontaneous explosion in a dry box caused 39 curies of polonium to be released; cost of accident, $1900	None
August 21, 1959	Aiken, S.C.		AEC	Hot liquid containing high-level radioactivity vaporized through an open connection; decontamination costs, $130,000	None
September 10, 1959	Ft. Worth, Texas		Licensee	While setting up a puff camera, a 28-curie Ir^{192} source was blown out of end of tube	Calculated exposures ranged from 800 mrem to 11 rem for six persons
October 1, 1959	Vallecitos, Cal.		Licensee	Radioactive sample holders were removed from a pool which was loaded with irradiated and nonirradiated samples of europium	One person received a calculated exposure of 70 to 100 rem to one hand and 100 to 300 mrem to the whole body; another person received a calculated exposure of 7 to 17 rem to one hand, 90 rem to the other hand, and 200 to 400 mrem to the whole body
October 16, 1959	Idaho Falls, Id.	Vol. 1, No. 3, p. 75	AEC	A nuclear reaction in an underground waste collection tank resulted from the accidental jolting of fissionable material into the tank	Seven persons received whole-body exposures ranging up to 6 rem; of these, two received skin exposures of 50 and 32 rem; 14 others received negligible exposures
October 27, 1959	Asusa, Cal.		Licensee	As the result of a malfunction during the loading of a sample into an irradiator containing a 10,000-curie Co^{60} source, the sample was released prematurely and dropped onto the irradiator	None
November 11, 1959	Oak Ridge, Tenn.	Vol. 1, No. 4, p. 85	AEC	Particles of radioactive elements, principally Ru^{106} and Rh^{106}, were accidentally released from a stack	None

Table VI-1 (continued)

Date	Location	Nuclear Safety Review Citation	Installation Owner	Nature of Incident	Extent of Exposure
November 20, 1959	Oak Ridge, Tenn.	Vol. 1, No. 3, p. 78	AEC	About 0.6 g of plutonium was blown out of a concrete cell; cost of cleanup, $350,000	No persons were directly affected by the explosion; in subsequent cleanup operations, several persons received sufficient exposure to plutonium to result in measurable excretion of 1% of the maximum permissible body burden
November 30, 1959	Miamisburg, Ohio		AEC	Routine monitoring of two employees servicing radioactive equipment disclosed exposure to polonium; detailed circumstances of exposure could not be determined	One person received 3.9 times the permissible body burden of Po^{210}, and another 2.05 times the permissible body burden of Po^{210}
November 30, 1959	Idaho Falls, Id.		AEC	Rupture in a reactor core section of an in-pile tube caused spread of highly contaminated liquid; cost for cleanup of contaminated area and replacement of damaged instruments, $4720	None
December 12, 1959	Aiken, S.C.		AEC	Radioactive particulate matter shaken loose from a burial box being transported by rail resulted in the spread of contamination along the railroad right-of-way within the site; cost for cleanup, $5200	None
December 18, 1959	Oak Ridge, Tenn.		Licensee	Mechanical failure of a hood during the preparation of material containing I^{131} resulted in release into work area	One person received calculated total integrated dose of 106 rem to the thyroid
December 21, 1959	Jamaica, N.Y.		Licensee	Employee exposed while dismantling and removing heavy mechanical apparatus from a glove box contaminated with Sr^{90}	One person received a film-badge exposure of 170 mrem beta (whole body), 13,250 mrem to hand (over 13-day period)
January 1960	Dayton, Ohio		Licensee	Exposure while making instrument calibration checks with a Co^{60} source	One person received a film-badge exposure of 4 rem beta over the period Dec. 15 to Dec. 29, 1959, and 23 rem beta over the period Oct. 6, 1959 to Jan. 5, 1960
January 29, 1960	Everett, Mass.		Licensee	Employee neglected to return source to shielded position	One person received a calculated exposure of 130 rem to one hand

Table VI-1 (continued)

Table VI-1 (continued)

Date	Location	Nuclear Safety Review Citation	Installation Owner	Nature of Incident	Extent of Exposure
March 8, 1960	Oak Ridge Operations Office	Vol. 2, No. 2, p. 73	AEC	When cleaning up a cell the monitoring system failed to disclose the presence of radioactive Ce^{144}	One person received a beta exposure of 5500 rad to his hand
March 10, 1960	Pleasanton, Cal.		Licensee	Exposure to gaseous I^{131} and I^{133} from a fission-gas bubble released from a defective loop element	One person received a thyroid dose of 48 rad; six others received a calculated thyroid dose from 1875 to 7060 mrem
March 14, 1960	Portsmouth, N.H.		Licensee	Malfunction of the camera using a 40-curie Ir^{192} source	One person received a film-badge exposure of 6515 mrem
March 29, 1960	Madison, Wis.		Licensee	Samples were removed from an irradiator in which a 200-curie Co^{60} source had become detached and had fallen into the flask holding the samples	One person received an estimated 80 to 100 rem dose; eight others received estimated doses of 0.2 to 25 rem
April 3, 1960	Pittsburgh, Pa.	Vol. 2, No. 2, p. 70	Licensee	Release of mixed fission products to the primary coolant of the reactor caused by rupture of fuel element	None
April 5, 1960	Savannah River Operations Office		AEC	A leak in an outlet nozzle on a reactor vessel that resulted in the loss of heavy water coolant; repair cost, chiefly for repairing the nozzle and replacing the heavy water, $216,285	None
April 10, 1960	Pittsburgh, Pa.		Licensee	Exposure occurred while making radiographic shots with a 5.2-curie Ir^{192} source in a faulty camera	One person received a film-badge exposure of 120 rem
April 17, 1960	Hanford Operations Office	Vol. 2, No. 4, p. 52	AEC	Fire and explosion of uranium metal in a chemical dissolver damaged dissolver and contaminated cell, canyon, and crane; extent of loss, $250,000	None
April 26, 1960	Oak Ridge Operations Office	Vol. 2, No. 1, p. 53	AEC	A change in air pressure inside a hot cell forced contaminated graphite dust from the cell, and it dispersed into the rest of the building; resulting contamination cleanup cost, $39,500	Twenty-one persons received some exposure, with the highest calculated to be 2.1 rem integrated over 1 yr
April 26, 1960	Richland, Wash.		AEC	An employee's right forefinger was pricked by a splinter of plutonium through the rubber glove he was wearing while arranging equipment in a hood	Survey indicated 90,000 disintegrations per minute in the wound

Table VI-1 (continued)

Date	Location	Nuclear Safety Review Citation	Installation Owner	Nature of Incident	Extent of Exposure
April 27, 1960	St. Louis, Mo.		Licensee	A leak occurred in a 174-curie Co^{60} sealed source contained in a teletherapy unit	None
June 2–5, 1960	In transit from Baltimore, Md., to Corning, N.Y.		AEC	Water dripped from the vent of the expansion tank of a transfer cask which contained fuel elements	None
June 2, 1960	Savannah River Operations Office		AEC	During shipment of irradiated fuel elements, 30 to 40 gal of contaminated water leaked from the cask; decontamination costs in the area, $24,000	None
September 13, 1960	Savannah River Operations Office		AEC	Contaminated cooling water discharged from canyon onto floor; large loss, principally from cost of decontamination, $250,000	None

until September 1959, it would not be expected that any incident prior to 1959 would have been reviewed as a feature in the Current Events section of the journal. (W. B. Cottrell)

Changes in Licensing Regulations

The recent reorganization of the AEC Division of Licensing and Regulation was reviewed in the previous issue of *Nuclear Safety*,[7] and two amendments to the *Code of Federal Regulations*, Title 10, Part 50, on construction and operating license requirements were cited. A number of other changes that have been made in these regulations, including an entirely new proposed regulation on site evaluation criteria, are reviewed below.

A new section of the AEC regulation on "Standards for Protection Against Radiation," Title 10, Part 20, which establishes criteria for containers for sea disposal of waste, was published[8] in August 1960, and comments were invited. In September 1960, another proposed amendment to Part 20, which deals with permissible dose levels and concentrations, was published for comment, along with the statement that it was to become effective on Jan. 1, 1961. A few comments had been received[9] by mid-November 1960; some individuals viewed the proposed rules as overly restrictive, particularly in reporting requirements. The receipt of comments continued,[10] including more complaints regarding reporting requirements. The amendment, as published on Dec. 30, 1960, included minor changes,[11] such as definitions of "calendar quarters" and "restricted areas," some changes in the form (AEC-3) for informing employees of hazards, and an explanation of dust concentration restrictions in uranium mills. A revision was published[11] on Jan. 18, 1961, which stated that no license will be issued for disposal of material received from AEC licensees for burial on land not owned by the federal or state governments.

Amendments to the regulation on "Licensing of By-product Material," Title 10, Part 30, which exempted specified isotopes from license requirements, were adopted[12] in August 1960. Further amendments published[13] in December 1960 set forth prelicensing evaluation procedures to determine compliance with the rules and to provide by-product material regulations. The requirements for exemption of tritium from licensing for use in watch-dial illumination were set forth in detail.

The regulation on "Control of Source Material," Title 10, Part 40, which governs licenses for the use of source materials, was published[14] in final form in January 1961 and was to be effective 30 days later. The final regulation is essentially the same as the proposal published in September 1960, except that it provides that licenses issued in connection with the operation of a facility may be issued for periods longer than three years. The new rules expand the number of persons exempt from licensing requirements and designate more explicitly who comes under the general licensing provisions. A broad class of finished products containing thorium or uranium is exempted from licensing requirements. A new rule defines source material as, "(1) uranium or thorium or any combination thereof, in any physical or chemical form, or (2) ores which contain by weight 0.05% or more of (i) uranium, (ii) thorium, or (iii) any combination thereof." Under the revision, many users of small quantities of uranium would be relieved of the requirement of obtaining a specific license, although the general license would still be required.

A standard transfer and receipting form for shipments of special nuclear material has been published[13] as an amendment to the regulation on "Special Nuclear Material," Title 10, Part 70.

The AEC has proposed[15,16] a new regulation on "Reactor Site Criteria," Title 10, Part 100, which sets up environmental criteria as guides for use in determining the suitability of sites for power and test reactors from the standpoint of hazards to the public, and has invited comments on the proposal. This proposed regulation is discussed in the article entitled "Reactor Siting Trends and Developments" in Sec. I of this Review. The criteria are to be considered only as guides because the determination of quantitative criteria is not yet sufficiently precise to eliminate the need for judgment by the designer, the Commission's technical staff, and the ACRS. It is pointed out in Section 100.10 that the environmental criteria requirements will naturally be somewhat dependent on the type and design of the reactor. It is stated that, if some of the site characteristics are unfavorable, "the proposed site may nevertheless be found to be acceptable if the design of

the facility includes appropriate and adequate engineering safeguards."

In order to attain the objectives of preventing or minimizing both somatic and genetic damage from credible accidents, the AEC proposes criteria for (1) an "exclusion area" controlled by the operator, (2) a "low population zone" in which the residents could be protected (by evacuation if necessary) in case of accident, and (3) a "population center distance" to the nearest boundary of a densely populated center containing more than 25,000 persons. The calculated maximum credible dosages for persons at the exclusion-area boundary and in the low-population zone are not to exceed a total-body exposure of 25 rem or an iodine exposure of 300 rem to the thyroid in a 2-hr period. Another proposed guide is that the population-center distance should be at least 4/3 as far as the outer boundary of the low-population zone from the reactor. In addition, the geological, meteorological, hydrological, and seismic characteristics would continue to be considered, as well as any other physical characteristics of the area, in judging suitability for a reactor site.

An appendix to the proposed regulation presents formulas for calculating the defined distances. Factors included are activity release from containment, meteorological parameters and formulas, and suggested minimum exclusion-area and population-center distances relative to reactor thermal power. Application of the guides to existing and proposed reactor installations shows[16] that practically all power reactors approved for construction meet the suggested minimum values.

In December 1960 the AEC proposed[17,18] a regulation (Title 10, Part 115) which would place its power demonstration reactors under formal hearing requirements. This would apply to AEC-owned reactors operated at non-AEC installations as part of the power-generation facilities of electrical utilities systems.[19] In line with this decision, AEC announced that a public hearing would be held on Feb. 1, 1961, on the N.S. *Savannah* (this hearing was later postponed to Mar. 6, 1961.

The liability and property-damage insurance requirements[20] for reactors specified in "Financial Protection Requirements and Indemnity Agreements," Title 10, Part 40, were reviewed[21] in June 1960, and the latest amendments have been in effect since May 1, 1960. The AEC invited continuing review and received conflicting comments from the operating and insurance industries. The insurance industry recommended that insurance requirements rise rapidly with reactor power at first (the "square root" relation), then transfer to the old (May 1, 1960) relation at higher powers. Several operators[22,23] objected to this, pointing out the already comparatively high cost and arguing that the risk does not rise that fast in relation to the power. A comment from Armour Research Foundation[24] suggested that the government bear part of the risk so that the rates for insurance would make operation more feasible.

(L. A. Mann)

Legislation and Organization

In an address[25] on the status of atomic energy in the United States, presented at the joint American Nuclear Society–Atomic Industrial Forum meeting at San Francisco (Dec. 14, 1960), Congressman Chet Holifield said that the Joint Committee on Atomic Energy is conducting a major study of the problem of complexity in today's licensing and regulatory procedures. He also urged getting safety questions "out on the table" and into public discussion "since economic power is tied to safety and in no little part to public acceptance," and he emphasized the importance of spending for safety in reactor projects as willingly as for lawyers and public relations experts. He urged the AEC to emphasize its neutrality and to avoid the appearance of advocating a particular interest in a safety contest in order to preserve public confidence in its safety findings.

During the same week, in meetings of the Atomic Industrial Forum, Berman and Hydeman, codirectors of the Atomic Energy Research Project at the University of Michigan Law School, recommended[26] dividing the AEC into two agencies responsible, respectively, for (1) promotion and operation and (2) regulation of atomic energy activities in order to avoid conflicting tendencies within one body. A similar thought was expressed in a recent letter from the ACRS to the AEC,[27] which stated, in part, that:

> There should be a separation at the highest level practicable between the responsibility for promotion and the responsibility for regulatory and safety activities. This separation might be obtained by having two separate agencies in the Executive

Branch of the Government, but the Advisory Committee on Reactor Safeguards should exist within the Commission itself. It is the view of the ACRS that a satisfactory solution could be obtained if:

A) a Commissioner were to concern himself with regulation and safety as his special sphere of interest, and

B) the line organization to implement safety review, inspection, licensing, and hearings were to be entirely separate from the promotional and developmental activities of the Commission.

Other suggestions regarding the AEC regulatory process were offered by Heimann,[28] Counselor, General Electric Company, for use in uncontested license cases to minimize time consumption without compromising safety responsibilities. He proposed that the staff present a report of its findings to the ACRS, who would have concurrently been studying the case, and that the ACRS then prepare and present its conclusions for use by the AEC Hearing Examiner, who could, of course, pursue any question he wished to investigate further. The Examiner's report would then be submitted to the Commissioners for appropriate action. Heimann also stressed the point that the need for standards will increase greatly with the increase in use of atomic energy, saying that work directed toward the development of standards should be pushed now. He also recommended separation of the regulatory function of the AEC and the establishment of a new agency.

In December 1960 the Federal Radiation Council invited[29] public comments on radiation protection standards for controlling human exposure from environmental sources, requesting that comments be submitted by Feb. 15, 1961. The Council repeated the necessity of judging benefits against risks for the greatest good to the public. The Council's studies have used the NCRP recommendations as a basis, initially directing attention to the nuclides I^{131}, Ra^{226}, Sr^{89}, and Sr^{90}. Also being studied is the advisability of a graded series of guides in which the magnitude of corrective action would increase as activity concentration in the environment increases.

The Department of Defense (DOD) recommended[30] legislation authorizing DOD to indemnify contractors against nuclear and other unusually hazardous risks on the basis that it will be unable to obtain contracts, in many cases, unless it can indemnify the contractors. DOD requested that the legislation cover a 10-year period and then be subjected to review. Safeguard clauses were recommended to prevent misuse of the authority. (L. A. Mann)

Action on Reaction Projects by Licensing and Regulating Bodies

The AEC is required by the Atomic Energy Act to protect the public from undue hazards resulting from exposure to radioactivity. In order to fulfill this responsibility, while furthering its simultaneous responsibility for developing the use of nuclear energy, the Commission has enacted and amended regulations and has published them in the *Code of Federal Regulations*. Title 10 of the Code describes two major steps that must be taken in order for a nuclear reactor to be built and operated by private interests: (1) securing a construction permit and (2) obtaining an operating license. Title 10 of the Code requires the AEC to make certain specific findings regarding the safety of the public before issuing either the construction permit or the operating license.

The AEC has used the technical findings and advice of the legislatively formed ACRS, which was set up by act of the U. S. Congress, as well as the advice and findings of its own Division of Licensing and Regulation, to assist in deciding when construction permits or operating licenses should be issued to private organizations and when approval should be given for construction or operation of government-owned nuclear facilities. Licensed reactors are subject to a public hearing before issuance of a construction permit or operating license.

The AEC has also set up regulations for the procedures and the reporting, etc., required to assure the continued safety of operation of nuclear facilities. Prior approval is required for any significant changes in equipment, operating level, or method of operation, and periodic inspections must be made by specified AEC personnel. The degree of detail and the frequency of the inspections depend on the judgment of the AEC relative to each facility. A significant development in the regulation of changes in facility equipment of operation by the AEC has been the extension of a certain freedom of judgment to the operator as to whether a contemplated change contains an element of hazard not previously reviewed and

approved by the Commission. The first relaxation in this direction was ordered in the case of Vallecitos Boiling-Water Reactor (VBWR), Nov. 2, 1960, and similar orders have since been issued for other operations, including the Westinghouse Test Reactor (WTR) and Yankee Nuclear Power Station operations (see specific case reviews below).

Several AGN-201 reactor licensees were notified Dec. 21, 1960, of nine new conditions to be added to their license requirements.[31] The conditions relate to fuel loading and fuel movement, dependability of control-rod action, and to instrumentation and controls requirements.

Early in January 1961, the AEC sent telegrams to 49 licensees requiring a report on specified items pertaining to reactor safety, especially in preventing inadvertent criticality.[32] The reactor operators were notified that the reactor fuel had to be on the premises and inspected before preoperational inspection could be considered complete.

As of Nov. 15, 1960, the facilities license application record[33] was that reported in Table VI-2. In addition, 39 export licenses had been issued, and two were pending.

The status of several power and test reactors in various stages of study, construction, and operation is given in Table VI-3. A table of tentative schedules for proposed reviews of application was listed in the previous issue of *Nuclear Safety*.[34] Some of the more interesting features of reactor and facility status are summarized below.

Big Rock Point Reactor (Docket 50-155)

The 240-Mw(t) forced-circulation boiling-water reactor for Big Rock Point was approved for construction in May 1960. This reactor is to be operated by Consumers Power Company. The previous issue of this Review[34] outlined its status and some proposals that were to be considered at a hearing scheduled for Dec. 13, 1960. At the request of Consumers Power Company, the hearing on the construction permit was postponed[35] "to await further reports and analyses of a control-rod-drive failure at another nuclear power reactor before presenting evidence" regarding the control-rod-drive mechanism proposed for the Big Rock Point reactor. Presumably the "other reactor" is the Dresden reactor, whose status is discussed below.

Dresden Reactor (Docket 50-10)

The Dresden reactor, a 630-Mw(t) boiling-water type of power reactor, was shut down[34] on Nov. 15, 1960, when difficulties with a control rod became apparent. Investigation revealed failure of a control-rod-drive tube caused by stress corrosion, and, when all the drive tubes were removed and inspected, cracks were found in other drive tubes. The reactor builder, the General Electric Company, decided that age-hardened Armco 17-4 stainless steel of the type originally used would be satisfactory for the drive tubes if given a higher temperature heat-treatment and that some alternate, more ductile, material should be used for parts subject to less severe wear and galling conditions.[36] It was concluded that all the Dresden drives would be rebuilt and that extensive tests would be made. On Jan. 30, 1961, the applicant (Commonwealth Edison Company, Chicago) submitted a report of the diagnosis, proposed modifications in the design and manufacture of the drives and in the test program, and re-

Table VI-2 STATUS OF LICENSE APPLICATIONS AS OF NOV. 15, 1960

Type of reactor or facility	Number of licenses in status shown				
	Construction permit pending	Permit issued	Application withdrawn	Operating license	
				Pending	Issued
Power	2	7			3
Test		1	2		2
Research		19	7	4	66
Critical experiment facilities		2			14
Production facilities	2				
Total	4	29	9	4	85

Table VI-3 CALENDAR OF LEGAL STEPS IN LICENSING U. S. POWER REACTORS

REACTOR INFORMATION							CONSTRUCTION PERMIT					OPERATING LICENSE					
CODE No.	NAME OPERATOR LOCATION	REACTOR TYPE	MAXIMUM POWER Mwt	MAXIMUM POWER Mwe	HAZARDS REPORT No.	HAZARDS REPORT DATE	FINDINGS OF ACRS-HEB or AEC STAFF	FINDINGS OF HEARING EXAMINER	CONSTRUCTION PERMIT No	CONSTRUCTION PERMIT DATE	SCHEDULED COMPLETION DATE	ADDITIONAL INFORMATION	FINDINGS OF ACRS-HEB or AEC STAFF	FINDINGS OF HEARING EXAMINER	OPERATING LICENSE No.	OPERATING LICENSE DATE	OPERATING LICENSE EXP	ADDITIONAL INFORMATION
PP-4	BONUS PUERTO RICO WATER RESOURCES AUTHORITY RINCON, PUERTO RICO	BOILING WATER NUCLEAR SUPERHEAT	50	17.3	PRWRA-GNEC-2	12-21-59	3-14-60 ACRS FAVORABLE 6-60 AEC FAVORABLE	6-28-60 FAVORABLE	NOT REQUIRED		9-62	CONSTRUCTION AUTHORIZED EFFECTIVE 7-19-60						
50-144	CVTR (CAROLINAS-VIRGINIA TUBE REACTOR) CAROLINAS-VIRGINIA NUCLEAR POWER ASSOCIATES, Inc. PARR, So CAROLINA	PRESSURIZED D_2O MODERATOR PRESSURIZED D_2O COOLANT	60.5	17	CVNA-27 (3 vols)	7-15-59	12-11-59 AEC FAVORABLE	4-12-60 FAVORABLE	CPPR-7	5-4-60	1-1-62 to 12-31-62							
50-155	BIG ROCK POINT CONSUMERS POWER Co BIG ROCK POINT CHARLEVOIX COUNTY, MICHIGAN	BOILING WATER	240	75	NO No	1-60	4-18-60 ACRS FAVORABLE	4-18-60 FAVORABLE	CPPR-9	5-31-60	7-1-62 to 7-1-63							
50-10	DRESDEN COMMONWEALTH EDISON Co GRUNDY COUNTY, ILLINOIS	BOILING WATER	630	192 gr 180 net	NO No. (4 vols) SUMMARY	THRU 9-57 1-6-59			CPPR-2	5-4-56		CONSTRUCTION STARTED MARCH, 1957 COMPLETED SEPTEMBER, 1959			DPR-2	9-26-59 6-9-60 10-14-60	12-10-59 5-4-96	($\frac{1}{2}$ POWER) * TEMPORARY (FULL POWER) TERM 40yr FULL POWER
PP-1	ELK RIVER ELK RIVER RURAL COOPERATIVE POWER ASSOCIATES ELK RIVER, MINNESOTA	BOILING WATER	58	22	NO No (revised) SUPP 1	3-12-59 5-25-59		1-19-60 FAVORABLE 4-60 FAVORABLE	NOT REQUIRED				11-5-60 ACRS FAVORABLE					HEARING SCHEDULED 3-7-61
F-16	FERMI (ENRICO FERMI FAST BREEDER) POWER REACTOR DEVELOPMENT Corp LAGOONA BEACH, MONROE COUNTY, MICHIGAN	FAST NEUTRON PLUTONIUM BREEDER SODIUM COOLED	300	104 gr 94 net	APDA-108 APDA-117 APDA-115 APDA-124 EXHIBITS THRU T-18	9-1-55 10-15-56 11-1-56 1-59 THRU 9-10-59			CPPR-4 REAFFIRMED REAFFIRMED	8-4-56 12-10-58 5-26-59	7-15-61	VALIDITY OF CONSTRUCTION PERMIT IS BEFORE THE UNITED STATES SUPREME COURT						
	EGCR (EXPERIMENTAL GAS COOLED REACTOR) TENNESSEE VALLEY AUTHORITY OAK RIDGE, TENNESSEE	HELIUM COOLED GRAPHITE MODERATED FUEL TESTING	84	25	ORO-198	5-19-59	5-59 ACRS FAVORABLE		NOT REQUIRED		12-31-62	CONSTRUCTION APPROVED AND IN PROGRESS						
50-152	FWCNG FLORIDA WEST COAST NUCLEAR GROUP POLK COUNTY, FLORIDA	D_2O MODERATED CO_2 COOLED	153	50	GNEC-115 (revised) (3 vols)	12-7-59	APPLICATION PENDING					RESTUDYING COSTS						
PP-3	HNPF (HALLAM NUCLEAR POWER FACILITY) CONSUMERS PUBLIC POWER DISTRICT HALLAM, NEBRASKA	GRAPHITE MODERATED SODIUM COOLED	240	75 gr	NAA-SR 3579 CPPD-28 NAA-SR 4504	NO DATE 9-22-55 NO DATE	7-25-59 ACRS FAVORABLE 2-8-60 ACRS FAVORABLE 4-29-60 AEC FAVORABLE	6-60 FAVORABLE	NOT REQUIRED		7-62	CONSTRUCTION AUTHORIZED EFFECTIVE 6-28-60						
50-133	HUMBOLDT BAY PACIFIC GAS & ELECTRIC Co HUMBOLDT BAY, CALIFORNIA	BOILING WATER VAPOR SUPPRESSION (SAFETY)	163	60 gr	NO No	4-15-59	7-60 ACRS FAVORABLE	10-17-60 FAVORABLE	CPPR-10	11-9-60	1-1-62 to 7-1-63							
50-3	INDIAN POINT CONSOLIDATED EDISON Co of NEW YORK BUCHANAN, NEW YORK	PRESSURIZED WATER THORIUM CONVERTER	585	275 gr 255 net	EXHIBITS THRU I-4 EXHIBITS THRU K-4 EXHIBIT K-5 (rev 1)	THRU 1955 THRU 1956 1-60			CPPR-1	5-4-56	10-1-60 to 10-1-61							
	N S SAVANNAH MARITIME ADMINISTRATION UNITED STATES DEP'T of COMMERCE	PRESSURIZED WATER	69	22 000 shp	BAW 1164	7-60			NOT REQUIRED			UNIQUE LEGAL STATUS PRESIDENTIAL RECOMMENDATION CONGRESSIONAL AUTHORIZATION						CRITICAL TESTS AND TRIAL RUNS SCHEDULED IN 1961
50-130	PATHFINDER NORTHERN STATES POWER Co. SIOUX FALLS, So DAKOTA	BOILING WATER NUCLEAR SUPERHEAT	203	66 gr 62 net	ACNP-5905 ACNP-5917	3-10-59 8-31-59	3-60 ACRS FAVORABLE	4-60 FAVORABLE	CPPR-8	5-12-60	1-1-62 to 7-1-62							
50-171	PEACHBOTTOM PHILADELPHIA ELECTRIC Co. CONOWINGO POND, YORK COUNTY, PENNSYLVANIA	HIGH TEMPERATURE UNCLAD FUEL GAS COOLED GRAPHITE MODERATED	115	40	NO No.	12-59	12-10-60 ACRS SITE FAVORABLE											
PP-2	PIQUA OMR CITY of PIQUA MUNICIPAL POWER COMMISSION PIQUA, OHIO	ORGANIC MODERATED ORGANIC COOLED	46	11.4	NAA-SR 3100 SP NAA-SR 3279 NAA-SR 3575 NAA-SR 4048	9-28-58 NO DATE 4-13-59 7-1-59	7-25-59 ACRS FAVORABLE	1-7-60 FAVORABLE				CONTINUANCE OF CONSTRUCTION AUTHORIZED EFFECTIVE 1-28-60						
	PWR DUQUESNE LIGHT Co SHIPPINGPORT, PENNSYLVANIA	PRESSURIZED WATER	231	68 gr 60 net	WAPD-SC-547 * WAPD-SC 541 OTHER SUPP	6-57 9-57 11-57	1-20-54 ACRS FAVORABLE					CONSTRUCTION STARTED 5-55	11-4-57 ACRS FAVORABLE					OPERATING AT FULL POWER SINCE 12-23-57 NEW SEED ELEMENT AUTHORIZED CRITICAL 4-12-60 ON LINE 5-6-60
50-146	SAXTON SAXTON NUCLEAR EXPERIMENTAL Corp BERKS COUNTY, PENNSYLVANIA	PRESSURIZED WATER	20	5 net	NO No.	NO DATE	1-7-60 ACRS FAVORABLE	1-21-60 FAVORABLE	CPPR-6	2-11-60	12-1-61 to 12-1-62						SEE NEXT COL	TERM TO 2-11-66 OR 5 YEARS AFTER CRITICALITY, WHICHEVER COMES EARLIER
50-18	VBWR (VALLECITOS BOILING WATER REACTOR) GENERAL ELECTRIC Co PLEASANTON, ALAMEDA COUNTY, CALIFORNIA	BOILING WATER TEST	50* 50 *	3	GEAP-089 SS-VAL-2	10-17-55 8-31-57			CPPR-3	5-14-56					DPR-1	8-31-57	5-14-63	OPERATING SINCE 9-57 * OPERATING AT 50 Mwt 50 Mwt AUTHORIZED
50-29	YAEC YANKEE ATOMIC ELECTRIC Co ROWE, MASSACHUSETTS	PRESSURIZED WATER	392	145 gr 136 net	NO No (2 vols)	NO DATE			CPPR-5	11-4-57			1-30-60 ACRS FAVORABLE 5-9-60 ACRS FAVORABLE	5-17-60 FAVORABLE	DPR-3	7-9-60 7-29-60		PROVISIONAL 5 Mwt AND FULL POWER TESTS CRITICAL 8-19-60

quested AEC evaluation of the proposals.[37] The applicant also requested concurrence of the AEC with the applicant's plans to resume operation of the reactor upon successful completion of a test program. The applicant also submitted, on Jan. 27, 1961, a proposed amendment to provide for installation and operation of 112 new fuel assemblies to replace those already in the Dresden reactor core.[37]

Elk River Reactor (Docket PP-1)

A hearing scheduled for Dec. 20, 1960, on the 73-Mw(t) Elk River boiling-water reactor was postponed on the grounds that fuel fabrication problems prevented following the original fuel-loading schedule.

Humboldt Bay Reactor (Docket 50-133)

On Nov. 10, 1960, the Pacific Gas and Electric Company submitted an amendment to its license application for the 163-Mw(t) Humboldt Bay boiling-water reactor to provide for the use of stainless-steel cladding for the fuel elements for the first core instead of the Zircaloy previously specified.[38] It was stated, however, that: "The Company expects to use zirconium-clad cores in the future, based upon what it anticipates will be shown by the further investigation of zirconium alloys now under way." The reason for the change was that tests had indicated possible difficulties with cladding imperfections in Zircaloy-clad elements.

Indian Point Reactor (Docket 50-3)

The applicant, Consolidated Edison Company of New York, submitted an amended application for license[39] of its 585-Mw(t) Indian Point pressurized-water thorium-converter reactor under construction at Buchanan, N. Y. The scheduled completion date has been amended to Oct. 1, 1961, from the previous date (Oct. 1, 1960). An estimated schedule of deliveries of special nuclear materials was presented.

PWR (Shippingport) Reactor

Operation of the PWR, a pressurized-light-water reactor rated at 231 Mw(t) and the first of the large power reactors, was reviewed in the September 1960 issue of *Nuclear Safety*,[40] based on reports of operation up to February 1960.

On Dec. 7, 1960, a résumé of the plant operation to date, including refueling, was presented at a meeting of the ACRS at the Shippingport site.[41] Proposed future operations were described, including the design and operational changes necessary for installing and operating core 2. With core 2, the power level can be increased to 505 Mw(t), and therefore major revisions in the design of the reactor and the heat-removal equipment are required. The ACRS stated[41] that the proposed modifications would not significantly alter the safety status of the facility but suggested some specific areas for development of improvements (e.g., leakage rate testing and a safety injection system).

Pathfinder Reactor (Docket 50-130)

On Nov. 4, 1960, the Northern States Power Company submitted[42] Amendment 7 to its license applications for the Pathfinder 203-Mw(t) boiling-water reactor that is to provide nuclear superheat. The amendment included "a detailed description of the waste-disposal systems, description of the reactor instrumentation and control system, and expanded coverage of the plant dynamic analysis."

On Jan. 5, 1961, the applicant reported that another amendment to the construction permit will be submitted for changes in fuel materials. Fourteen spare fuel elements and 13 "spike" elements will be ordered for adjustment and optimization of the first core. A "sizable rejection rate" is anticipated for superheater fuel elements for the first core, requiring a large withdrawal for superheater fuel. The previous issue of *Nuclear Safety*[34] presented ACRS comments regarding fuel-element cladding for this reactor.

Peach Bottom Reactor (Docket 50-171)

The ACRS reviewed, at its meetings of Dec. 7–10, 1960, the status of the Peach Bottom 115-Mw(t) high-temperature unclad-fuel gas-cooled reactor application and expressed[43] guarded optimism:

> The Committee continues to be optimistic that a reactor of the general type proposed in the advanced gas-cooled concept can be constructed and operated at the Peach Bottom site without undue risk to the health and safety of the public. General Atomic has embarked on an extensive research and development program related to this reactor. The initial results from this program appear to be favorable.

However, there are many questions that remain to be answered by the research and development program which is still in its early stages. Several of these questions are in areas which could require major changes in the present design concepts and could conceivably change our early optimism. For example, the long-term integrity of the graphite under the proposed design conditions, which lie far outside past experience with graphite, has yet to be established. Also, the experimental data required to determine that the fission products emitted continuously from the homogeneous fuel components can be collected, stored, and disposed of safely have yet to be developed. A novel design is proposed for the control-rod system, but the development is not yet sufficiently advanced to permit adequate evaluation of its reliability. The applicant has not provided either a secondary back-up safety system or an emergency coolant system but has not yet established to the satisfaction of the Committee that these systems are unnecessary. The relatively large assumed Doppler coefficient has yet to be confirmed.

In view of the foregoing, the Advisory Committee on Reactor Safeguards is not now prepared to go beyond its original conclusion reached in our letter to you under date of March 14, 1960, that the Peach Bottom site is suitable for a reactor of this general design and power level.

Plum Brook (NASA) Reactor Facility (Docket 50-30)

The Plum Brook Research Facility[34,44] includes a 60-Mw(t) light-water-cooled and -moderated reactor with a primary beryllium reflector and a secondary water reflector and is for use primarily in the development of "nuclear propulsion and nuclear power generation." On Nov. 15, 1960, the National Aeronautics and Space Administration (NASA) requested[45] a one-year renewal of the construction permit to Jan. 1, 1962, to allow time for additional work in four specified areas. This delay was occasioned by a shortage of qualified personnel.

At a Dec. 26, 1960, hearing conducted by J. D. Bond, AEC Hearing Examiner, NASA presented a proposal[46] for an 18-month provisional operating license for power levels up to 100 kw(t). The proposal was prepared in conjunction with the AEC staff. It provides a change procedure very similar to that granted[45] for the VBWR (Docket 50-18) on Nov. 2, 1960. Among the specific restrictions included in the proposal were that (1) any request for operation at powers exceeding 100 kw(t) would have to be referred to the ACRS and to a public hearing; (2) within 30 days after completion of the specific low-power test, NASA would have to file a report describing changes in facility design, performance characteristics, and operating procedures; (3) the initial core loading and later operation at powers exceeding 1 kw(t) would be permitted only in case of specified affirmative findings for those operations by the Director of the AEC Division of Licensing and Regulation. NASA indicated target dates of Feb. 15, 1961, for fuel loading and Aug. 15, 1961, for completion of construction for full-power operation of the reactor.[46]

Radiation Effects Reactor (Docket 50-172)

Tests have been made with the pressurized-water, water-cooled and -moderated, 10-Mw(t) test reactor facility [Radiation Effects Reactor (RER)] of the Georgia Nuclear Laboratories[47] for the U. S. Air Force by Lockheed Aircraft Corporation under a contract arrangement since the summer of 1958. Upon indication by the Air Force that it would lease the facility to Lockheed for work for other government agencies and for commercial and educational organizations after Jan. 1, 1961, Lockheed applied for appropriate licenses to carry out such work. The ACRS considered Lockheed's application at meetings held Dec. 7–10, 1960, and reported[48] the following findings:

> At this meeting, the Committee considered the limited operating performance of this facility over the past seventeen months. A revised contractual arrangement was described within which Lockheed will lease RER from the Air Force, rather than operating it under an Air Force R&D contract. Military requirements have diminished to the point where the contractor proposes to operate the reactor commercially, providing bulk radiation testing for other governmental organizations and private manufacturers. The Lockheed Corporation has applied for a facility license and will obtain AEC licenses for its operators as a result of this change.
>
> In its letter of July 25, 1959, to you, the Committee expressed concern regarding the instrumentation and lack of operator experience at this facility. These areas appear now to have been satisfactorily resolved. Previously this reactor was considered under Section 91b as a facility to be operated for military purposes. At the present time the Committee is being asked to consider this reactor for a civilian license so that it can be operated for commercial purposes. It is the opinion of the Advisory Committee on Reactor Safeguards that continued

operation of this reactor can only be justified for work essential to the national defense.

On Jan. 28, 1961, Lockheed advised[49] that, in view of the above ACRS recommendation, it "has no objection to receiving such license which allows operation of the RER for government-supported programs only." Contract AF 33(600)-42486 between the Air Force and Lockheed, for standby maintenance and operation of the facility, covers the period[50] Jan. 1, 1961, through Sept. 30, 1961.

Nuclear Merchant Ship N.S. Savannah

The N.S. *Savannah*[51,52] is to be powered by a 69-Mw(t) pressurized-water reactor. It employs a containment system that is necessarily different from that of a stationary power plant. Because of the unusual requirements for the power plant, the ACRS has devoted comparatively more time to review the N.S. *Savannah*, having made four progress reports on hazards studies to the AEC.[53] A hearing on the N.S. *Savannah*, the last before reactor criticality, was held on Mar. 6, 1961, after having been postponed from Feb. 1, 1961, "to permit substantial completion of the nonnuclear test program."[54]

The ACRS, in its Dec. 13, 1960, report[55] to the AEC, concluded that, with two restrictions, the reactor can be operated without undue hazard to the health and safety of the public. The two restrictions were that (1) the reactor power be restricted to 7 Mw(t) in the startup tests at Camden and (2) after further tests at higher powers (probably in the York River estuary) and subsequent limited sea trials, "the performance of the N.S. *Savannah* be documented for formal review by the staff and the ACRS" before returning the ship to Camden with the primary system pressurized or before starting the extended sea trials (phase VI testing).

> The Committee [ACRS] wishes to emphasize that this opinion [recommendation] does not reflect a lack of confidence in the design, start-up procedures, or operation of this reactor. It merely recognizes that, with any reactor which has not been extensively tested at power or as a full prototype, there exists a remote possibility of errors, in design or in operating procedures, which might be hazardous to the public.

The ACRS also recommended more study and some additional tests on control rods and drives, reactor compartment flooding in an emergency, and monitoring procedures for checking iodine and particulate filters.

Vallecitos Test Reactor (Docket 50-18)

The status of the boiling-water test reactor at Vallecitos was reviewed in the previous issue of *Nuclear Safety*,[34] including the Nov. 2, 1960, decision of the AEC to permit General Electric "complete freedom to make changes within the parameters of the technical specifications, provided no unresolved safety question is involved." In accordance with that decision, General Electric was advised[56] on Dec. 1, 1958, to review the Report of Proposed Changes, No. 4, to ascertain which of the changes described may now be made without prior AEC approval and to advise the Commission of each change made, along with a statement of the reasons why General Electric believes the change does not involve an unreviewed safety question. In this way this important precedent-setting decision to allow such freedom was put into action. Changes made by General Electric[57,58] without prior approval of the AEC were reported on Dec. 27, 1960, and Jan. 9, 1961. The decision has served as a precedent to similar agreements with other reactor owners (e.g., the Yankee and WTR operators).

Westinghouse Test Reactor (Docket 50-22)

The status of the Westinghouse 60-Mw(t) water-cooled tank type test reactor was reviewed in the March 1960 issue of *Nuclear Safety*[59] and again following the incident in which fuel-element melting occurred.[60] On Dec. 9, 1960, Westinghouse submitted[61] a request for a proposed amendment to its operating license TR-2. In this request, Westinghouse, in addition to repeating its position stated in its letter of Oct. 3, 1960:

> ... that change procedures, technical specifications and definitions of plant portions subject to regulation should be determined by the Commission in a rule-making procedure rather than on a case-by-case basis, ...

included in the proposed amendment the main features summarized below.

The amendment is described by Westinghouse as necessary and proper to "harmonize" generally with a similar action taken on Nov. 2, 1960, with respect to the VBWR (Docket 50-18)

(see preceding discussion). Westinghouse proposes to make changes in the facility or in operating procedures and to conduct tests and experiments which Westinghouse determines are not "unreviewed safety questions," as defined in the proposal. Any such changes made are to be reported and justified to the AEC on a quarterly basis. Changes which are "unreviewed safety questions" will continue to be submitted to AEC and will not be effected until approved by AEC.

Yankee Reactor (Docket 50-29)

The status of the Yankee Atomic Electric Company (YAEC) 392-Mw(t) pressurized-water reactor was reviewed in the previous issue of *Nuclear Safety*,[34] especially with regard to change procedures. A subsequent proposal by Yankee for a license amendment on change procedures was approved by the AEC staff.[62] A set of "technical specifications" developed by Yankee was included as an appendix to the proposal, being described as those specifications which were not to be changed without AEC authorization, with the Division of Licensing and Regulation to decide for each pertinent change proposed whether a public hearing would be required. The specifications mentioned purport to cover the areas that might involve nuclear safety. In other areas, Yankee was to be free to make changes without the requirement of AEC approval. Some differences from the VBWR amendment discussed above were described. On Dec. 30, 1960, AEC Hearing Examiner Jensch recommended general adoption of the proposed amendment.[63] The amendment was issued[64] to Yankee on Jan. 6, 1961.

(L. A. Mann)

Safeguards Reports and Selected Reading

The recently issued safeguards reports and selected literature pertaining to hazards and safety of reactors are listed below for reference.* Because of the similarity of many reactors (in particular, research reactors), this list is not intended to be all-inclusive.

*Some of these reports are not yet available, but they may be examined in the AEC's Public Document Room, Washington, D. C.

1. C. B. Graham, Final Safeguards Report, Allis-Chalmers Critical Facility, USAEC Report ACNP-5910, Allis-Chalmers Mfg. Co., Atomic Energy Div., May 1, 1959.

2. Atomics International, Hazards Summary Report for the Walter Reed Army Medical Center Nuclear Research Reactor, Report AI-3739(Suppl. 1), Dec. 1, 1959.

3. Atomics International, Hazards Summary Report, L-77 Laboratory Reactor, Report AI-Memo-2387(Suppl. 2), Mar. 11, 1960.

4. E. M. Reiback (Ed.), Hazards Summary Report — Prepackaged Nuclear Power Plant for an Ice Cap Location (PM-2A), Report APAE-49(Suppl. 1), Alco Products, Inc., Apr. 15, 1960.

5. General Atomic Div., General Dynamics Corp., Hazards Report for Torrey Pines TRIGA Reactor, Report GA-722, May 27, 1959.

6. Puerto Rico Water Resources Authority and General Nuclear Engineering Corp., Boiling Nuclear Superheater (BONUS) Power Station, Preliminary Design Study and Hazards Summary Report, USAEC Report TID-8524, four volumes, June 1960.

7. Westinghouse Electric Corp., Bettis Atomic Power Laboratory, PWR Hazards Summary Supplement, USAEC Report WAPD-PWR (RD-1)-596, October 1959.

8. J. R. Brown et al., Hazards Summary Report for the HTGR Critical Facility, USAEC Report GA-1210, General Atomic Div., General Dynamics Corp., Feb. 11, 1960.

9. L. R. Amyot et al., Safeguards Report for the MGCR Critical Experiment, USAEC Report GA-1100, General Atomic Div., General Dynamics Corp., Dec. 18, 1959.

10. Nuclear Laboratory, Convair (Fort Worth), a Division of General Dynamics Corp., Hazards Summary Report on a Training Reactor for Texas Technological College, Report FZK-119.

11. R. S. Harding and M. Slater, Advanced Critical Experiment Safeguards Report (NUSU Phases B and C), Report CEND-105(Suppl. 2), Combustion Engineering, Inc., Nuclear Div., Mar. 17, 1960.

12. R. M. Ball and A. L. MacKinney, Application for Amendment 5 to AEC License CX-1 To Permit Critical Experiment Tests with Cores Produced for Use with the Nuclear Merchant Ship *Savannah*, Report BAW-III, Babcock & Wilcox Co., Nov. 30, 1959.

13. J. L. Meem, University of Virginia Reactor: Description and Operation, Report UVAR-8, Amendment 2, Sept. 22, 1958.

14. University of Virginia Reactor: Description and Operation, Report UVAR-8, Amendment 3, Feb. 5, 1960.

15. Dick Duffey et al., Safeguards Evaluation of the University of Maryland Reactor, University of Maryland and Allis-Chalmers Mfg. Co., Report UMNE-1, Feb. 7, 1960.

16. R. Schamberger and G. Foster, Pawling Lattice Test Rig Hazards Summary Report, Report NDA-2109-3, Nuclear Development Associates, Inc., Sept. 30, 1959.

17. Martin Company, Nuclear Div., Preliminary Hazards Report for the LFBR Critical Experiment, Report MND-LFBR-2179, December 1959.

18. Vitro Engineering Co., Preliminary Hazards Report to the AEC for the Cornell University Dual Core Reactor, Report KLX-1789, Jan. 17, 1958.

19. Vitro Engineering Co., Intermediate Hazards Report to the AEC for the Cornell University Swimming Pool Reactor, Report KLX-1789, Vol. 1, Add. 1, Dec. 15, 1959.

20. Vitro Engineering Co., Intermediate Hazards Report to the AEC for the Cornell University Zero Power Reactor, Report KLX-1789, Vol. 2, Add. 1, Dec. 15, 1959.

21. General Atomic Div., General Dynamics Corp., Application to USAEC for Construction Permit and Utilization Facility License for the FLAIR Reactor to be Constructed at the TRIGA Facility of the John Jay Hopkins Laboratory, San Diego, California. Enclosure 2: Hazards Summary Report, Report GA-1263, Mar. 1, 1960.

22. R. J. French et al., Safety Report for the Carolinas-Virginia Tube Reactor Critical Experiments, Report WCAP-1368, Westinghouse Electric Corp., Atomic Power Dept., Dec. 21, 1959.

23. M. R. Scheve, Final Report, Task 4.00, Liquid Fluidized-Bed Reactor Experiment, USAEC Report MND-LFBR-2337, Martin Co., Nuclear Div., April 1960.

24. J. D. Simpson, Summary Hazards Evaluation of the SMR Facility, General Electric Co., Aircraft Nuclear Propulsion Dept., USAEC Report APEX-536, December 1959.

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1. M. K. Harmon (Comp.), Redox Dissolver Incident: Comprehensive Report, USAEC Report HW-66850, Hanford Atomic Products Operation, Sept. 13, 1960. (Classified)

2. H. P. Simonds, Multipurpose Dissolver Information Manual, USAEC Report HW-64172, Hanford Atomic Products Operation, Mar. 4, 1960.

3. D. F. Hayes, A Summary of Accidents and Incidents Involving Radiation in Atomic Energy Activities, June 1945 Through December 1955, USAEC Report TID-5360, Aug. 1, 1956, and supplements.

4. A Summary of Transportation Incidents in Atomic Energy Activities, 1949–1956, USAEC Report AECU-3613, December 1957.

5. Atomic Energy Commission Annual Report to Congress for 1960.

6. *Code of Federal Regulations*, Title 10, Part 20, Standards for Protection Against Radiation, 1958.

7. Organization and Procedures in Licensing and Regulation, *Nuclear Safety*, 2(3): 61–63 (March 1961).

8. *Atomic Industry Reporter*, 6(272): 262–263 (Aug. 17, 1960).

9. *Atomic Industry Reporter*, 6(285): 366–367 (Nov. 16, 1960).

10. *Atomic Industry Reporter*, 6(291): 411 (Dec. 28, 1960).

11. *Atomic Industry Reporter*, 7(295): 26 (Jan. 25, 1961).

12. *Atomic Industry Reporter*, 6(273): 267 (Aug. 24, 1960).

13. *Atomic Industry Reporter*, 6(289): 395 (Dec. 14, 1960).

14. *Atomic Industry Reporter*, 7(294): 20–22 (Jan. 18, 1961).

15. *Federal Register*, 26: 1224–1226 (Feb. 11, 1961).

16. *Atomic Industry Reporter*, 7(298): 51–52 (Feb. 15, 1961).

17. *Atomic Energy Clearing House*, 6(49): 1–2 (Dec. 5, 1960).

18. *Atomic Energy Clearing House*, 7(6): 25 (Feb. 6, 1961).

19. *Atomic Industry Reporter*, 6(288): 387–388 (Dec. 7, 1960).

20. *Atomic Industry Reporter*, 6(280) (Oct. 12, 1960).

21. Changes in Financial Protection Regulations, *Nuclear Safety*, 1(4): 13–16 (June 1960).

22. *Atomic Energy Clearing House*, 6(48): 23–24 (Nov. 28, 1960).

23. *Atomic Industry Reporter*, 6(287): 384 (Nov. 30, 1960).

24. *Atomic Energy Clearing House*, 6(52): 16–17 (Dec. 26, 1960).

25. *Atomic Energy Clearing House*, 6(51): 3 (Dec. 19, 1960).

26. *Atomic Energy Clearing House*, 6(52): 18–20 (Dec. 26, 1960).

27. Letter from L. Silverman to Hon. John A. McCone, Dec. 13, 1960, subject: Regulation and Safety of Nuclear Reactors.

28. *Atomic Energy Clearing House*, 6(52): 20–22 (Dec. 26, 1960).
29. *Atomic Energy Clearing House*, 6(51): 23–25 (Dec. 19, 1960).
30. *Atomic Energy Clearing House*, 7(2): 3 (Jan. 9, 1961).
31. *Atomic Energy Clearing House*, 7(1): 4–5 (Jan. 2, 1961).
32. *Atomic Energy Clearing House*, 7(3): 11–12 (Jan. 16, 1961).
33. *Atomic Energy Clearing House*, 6(48): 18–22 (Nov. 28, 1960).
34. Action on Reactor Projects by Licensing and Regulating Bodies, *Nuclear Safety*, 2(3): 63–72 (March 1961).
35. *Atomic Energy Clearing House*, 6(51): 10 (Dec. 19, 1960).
36. *Atomic Energy Clearing House*, 7(2): 5–6 (Jan. 9, 1961).
37. *Atomic Energy Clearing House*, 7(7): 5–6 (Feb. 13, 1961).
38. *Atomic Energy Clearing House*, 6(45): 4 (Dec. 5, 1960).
39. *Atomic Energy Clearing House*, 6(51): 10 (Dec. 19, 1960).
40. Shippingport Operating Experience, *Nuclear Safety*, 2(1): 102–107 (September 1960).
41. Letter from L. Silverman to Hon. John A. McCone, Dec. 13, 1960, subject: Shippingport Pressurized Water Reactor.
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43. Letter from L. Silverman to Hon. John A. McCone, Dec. 10, 1960, subject: Peach Bottom Atomic Power Station.
44. National Aeronautics and Space Administration, Final Hazards Summary Report for NASA Plum Brook Reactor Facility, Parts I–III, December 1959.
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47. Nuclear Reactors Built, Building, or Planned in the United States as of December 31, 1960, USAEC Report TID-8200(3rd Rev.), p. 17.
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50. *Atomic Energy Clearing House*, 7(7): 7 (Feb. 13, 1961).
51. Babcock & Wilcox Co., 1960. (Unpublished)
52. Action on Reactor Projects by Licensing and Regulating Bodies, *Nuclear Safety*, 2(2): 81–89 (December 1960).
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59. Activities of the Advisory Committee on Reactor Safeguards, *Nuclear Safety*, 1(3): 80–81 (March 1960).
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62. *Atomic Energy Clearing House*, 6(51): 28–33 (Dec. 19, 1960).
63. *Atomic Energy Clearing House*, 7(1): 21–23 (Jan. 2, 1961).
64. *Atomic Energy Clearing House*, 7(4): 6 (Jan. 23, 1961).

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